Industry Information on Fuel Storage Rack Criticality Analyses

Everett Redmond II, Ph.D.
Nuclear Energy Institute

May 1, 2009

NUCLEAR
ENERGY
INSTITUTE

NEI Guidance Document

- NEI and industry are preparing a guidance document for fuel storage criticality analyses
- This document will discuss the technical content of fuel storage criticality license amendment requests

NEI

NEI Guidance Document

- The goal is to standardize the content of future license amendment requests
- This will provide stability and improve the efficiency of the review process
- The guidance document will be provided to NRC for review, comment, and possible endorsement

Agenda

- Precedent and significant figures
- Guidance document
- BWR criticality analysis
- CASMO
- Application of ANSI/ANS 8.27
- Margin versus conservatism
- Fuel assembly misloading
- Reactivity effects of boraflex degradation

NEI

Precedent

- Currently an approved topical report is not available for new criticality analysis for fuel storage
- However, previous licensing basis forms a precedent
- Analyses at other plants may also form a precedent (e.g. codes)

10 CFR 50.68

- "... if flooded with borated water, and the k-effective must remain below 1.0 (subcritical)"
- "Subcritical" does not have a specific number of significant digits associated with it
- Industry has not viewed 1.0 or 0.95 in terms of two significant digits

10 CFR 50.68

- 0.995 – 0.999 would be acceptable values to industry
- 0.995 – 0.999 should be acceptable to NRC
- Treating 1.0 as two significant digits is not consistent with the treatment of other numerical values in 10 CFR 50 (e.g. 10 CFR 50.2 definition of low enriched uranium fuel, … less than 20%)

NEI

Topics for Inclusion in a 10 CFR 50 Fuel Storage Criticality Analysis Guidance Document

May 1, 2009

Nuclear Energy Institute

Purpose

- Discuss topics to be covered in guidance document
- Document would be issued through NEI to assist licensees in LAR preparation
- Generally topics, few proposed technical details for each topic

Outline

- Contributors
- Applicable regulations, standards, guidance
- Computer code methods
- New fuel vault models
- Spent fuel pool models – PWR and BWR
- Depletion calculations
- Fuel assembly storage limits

Outline (continued)

- Soluble boron credit
- Other credits in storage
- Modeling of rack absorber material
- Precedent and references
- Independent technical review of LAR

Contributors

Vendors	Utilities
Westinghouse	Entergy
GNF	Duke
Holtec	Exelon
AREVA	TVA
NETCO	
Nuclearconsultants.com	

Applicable Regulations

- 10 CFR 50.68
- 10 CFR 70.24 (if applicable)
- 10 CFR 50 Appendix A – GDC 62

Applicable Standards

- ANSI/ANS 8 series
  - 8.1 (NCS Outside Reactors)
  - 8.7 (Storage of Fissile Mat.)
  - 8.17 (LWR Fuel Outside Rx)
  - 8.21 (Fixed Absorbers)
  - 8.24 (Validation of Methods)
  - 8.27 (Burnup Credit)
- ANSI/ANS 57 series
  - 57.1 (LWR Fuel handling)
  - 57.2 (LWR Spent Fuel Storage)
  - 57.3 (LWR New Fuel Storage)

Applicable Guidance

- Kopp memorandum
- NUREG/CR-6665 (Depletion Conditions)
- NUREG/CR-6683 (Fresh Fuel Equivalencing)
- NUREG/CR-6698 (Validation)
- NUREG/CR-6801 (Axial Burnup Profile)
- NUREG-0800 Sections 9.1.1 and 9.1.2
- Approved methodology topical report

Computer Code Methods

- Monte Carlo code
  - KENO, MCNP, etc.
- Benchmarking to critical experiments
  - Both code and cross section library
  - Statistical and trend analysis
  - Area of applicability
  - Normality
  - Results of analysis available to NRC

Computer Code Methods

- Depletion (lattice) code
  - PHOENIX, PARAGON, CASMO, TGBLA, etc.
- Modeling
- Depletion uncertainty
  - 5% of reactivity decrement
  - Lower values if justified
- Cross section libraries

Computer Code Methods

- Deterministic codes used for reactivity determinations (e.g. differential calcs)
- Modeling
  - Area of applicability and limitation or conditions on use of code
- Benchmarking
  - Sufficient to ensure accuracy of differential calculations

New Fuel Vault Models

- Nominal models
- Fuel and rack manufacturing tolerance calculations
- Abnormal conditions
  - Eccentric loading, unless treated as tolerance
  - Non-channeled fuel for BWR analyses
- Accident considerations
  - Flooding and optimum moderation
  - Misloaded/misplaced assembly

Spent Fuel Pool Models

- Document may cover BWRs and PWRs in separate sections
- Similar areas covered together in this presentation

Spent Fuel Pool Models

- Nominal models – PWR
  - Assembly design selection discussed below
- Nominal models – BWR
  - Accounts for most reactive lattice
  - Can use 2D Monte Carlo models to conservatively eliminate leakage

Spent Fuel Pool Models

- Fuel and rack manufacturing tolerance calculations
- Examples
  - Fuel enrichment and density
  - Cladding thickness, pellet diameter
  - Pitch of rack cell, fuel rods
  - Storage cell size and wall thickness
  - Others may be included in guidance document

Spent Fuel Pool Models

- Discuss effect on tolerance calculations of:
  - Depletion
  - Decay time
  - Soluble boron
  - Integral absorbers
  - Rack absorber degradation and/or gaps

Spent Fuel Pool Models

- Exposure uncertainties
  - Depletion uncertainty 5% of reactivity decrement
  - Reactor record assembly burnup uncertainty covered later
- Spent fuel pool temperature
  - Account for most reactive nominal temperature
- Combination of biases and uncertainties
- Region, configuration, and rack interfaces

Spent Fuel Pool Models

- Abnormal conditions
  - Eccentric loading, unless treated as tolerance
- Additional abnormal conditions – BWR
  - Non-channeled fuel
  - Channel bulge

Spent Fuel Pool Models

- Accident considerations
  - Integrate double contingency principle
  - Dropped assembly (vertical or horizontal)
  - Misloaded/mislocated assembly
  - Boron dilution if soluble boron credited – PWR
  - Temperature beyond nominal range

Spent Fuel Pool Models

- Limited cell configurations
  - Administrative controls
  - Physical cell blocking devices
- In-containment fuel storage analyzed with same techniques

Depletion Calculations

- Selection of limiting assembly or lattice
  - BWR analyses account for most reactive bundle at most reactive time in life
  - PWR analyses may consider only lattice which is limiting at conditions of interest

Depletion Calculations

- Core operating conditions
  - Moderator temperature
  - Fuel temperature
  - Moderator density
  - Soluble boron concentration
  - Specific power
  - Burnable absorbers (BPRAs/WABAs/IBAs)

Depletion Calculations

- Axial burnup profile – PWR
  - Burnup shape(s) for burnup range credited
  - Impact of axial fuel zoning
  - Nodalization
- Decay time

Fuel Assembly Storage Limits

- Calculation of target k_{eff}
  - Maximum calculated k_{eff} for storage
- Determination of required minimum burnup
  - Burnup that equals target k_{eff}
- Fitting of limits
  - Enrichment function
  - Decay time function

Fuel Assembly Storage Limits

- k_{inf} in standard cold core conditions – BWR
- Assembly burnup (reactor record) uncertainty treatment options
  - Uncertainty in target k_{eff}
  - Bias to burnup limit curve
  - Applied by site in determination of compliance

Soluble Boron Credit

- Normal operating conditions
- Accident considerations
  - Accident scenarios discussed above
- Determination of required concentration
  - Direct simulation of required concentration
  - Conservative worth curve determination

Other Credits in Storage

- Fresh integral BAs
  - NFV and/or SFP
- Spent fixed BAs
  - WABA or BPRA as water displacer
- Control rods
- Borated inserts

Modeling of Rack Absorber Material

- Dimensions and composition
  - Width
  - Length
  - Thickness
  - Density
- Modeling degraded absorbers
  - Gap sizes and distribution

Precedent and References

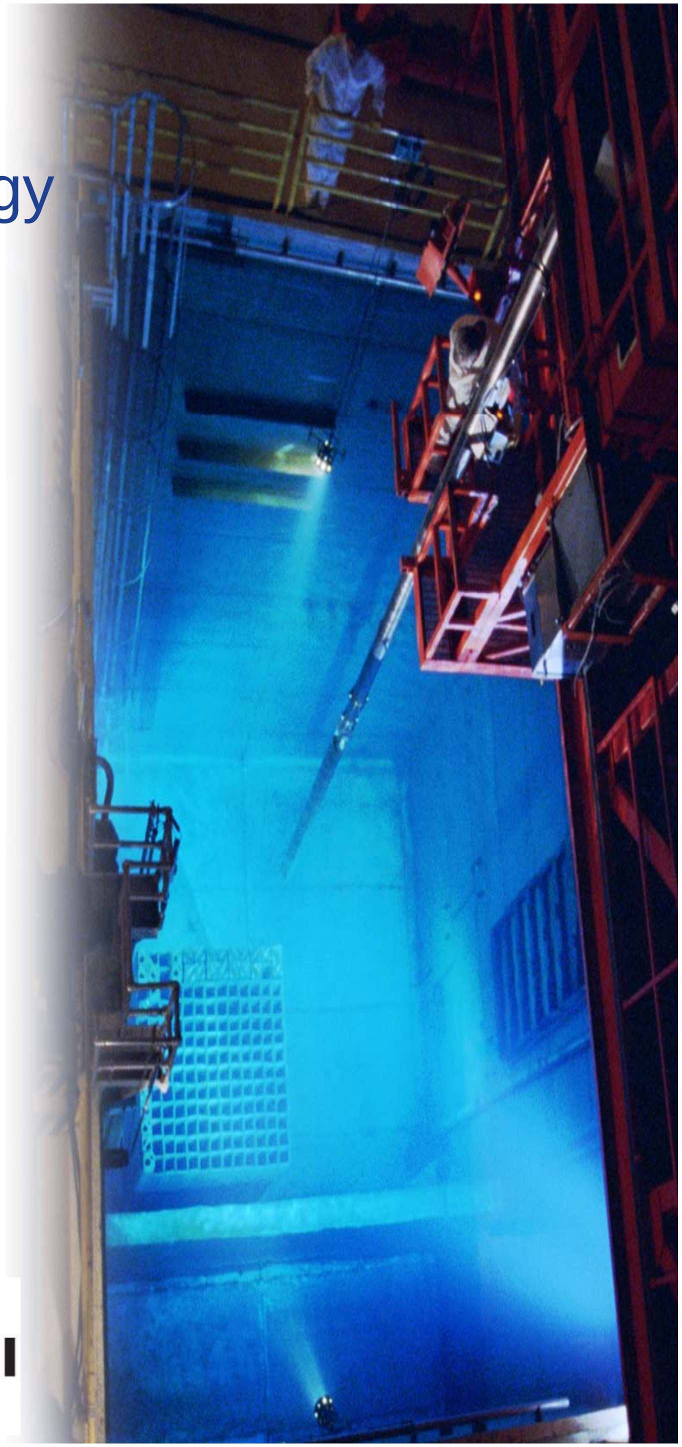
- Precedents can be used
- Differences should be identified and addressed
- Similarities should be highlighted
- Allows evaluation of applicability of precedent

Preparation of LAR

- NEI 06-02 provides guidance for preparation
- Licensee verifies completeness and accuracy of LAR
- Ensure high quality document is submitted

Global Nuclear Fuel

GE Hitachi
Nuclear Energy



Criticality Analysis of Nuclear Fuel Storage Racks

The BWR Story

Walid Metwally and Webb Mills

May 1, 2009
NRC

Global Nuclear Fuel
A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

Topics

- Activities at GNF/GEH
- Storage racks and the overall analysis process
- Storage rack modeling
- Conservative assumptions
- Criticality analysis and results

Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

Related Activities at GNF/GEH

- GE once manufactured Low-density and high-density fuel racks.
- GEH manufactures dry casks (no licensing).
- Perform criticality safety analyses for:
    - fresh and spent fuel racks,
    - fuel handling,
    - fuel shipping casks, and
    - fuel manufacturing.

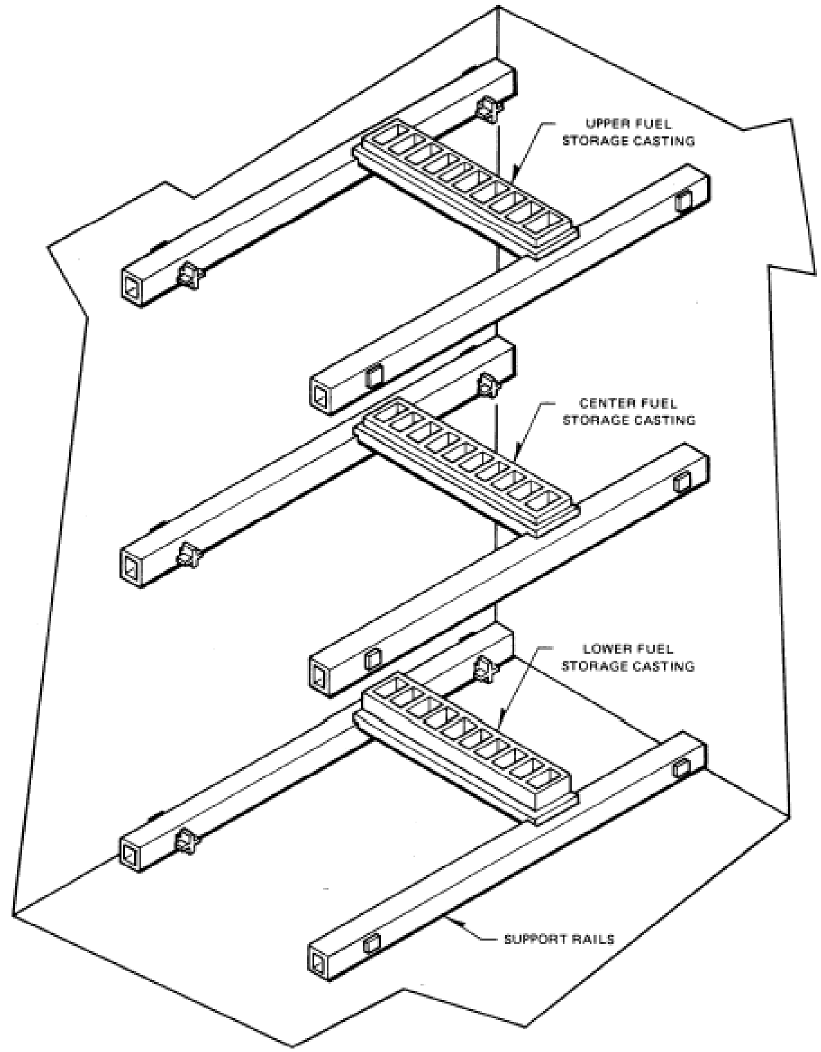
Fuel Storage Racks

- Low Density Fuel Storage (LDFS)
  - Non-poisoned
  - Fresh (new) fuel storage
  - Containment building fuel storage

- High Density Fuel Storage (HDFS)
  - poisoned
  - Spent fuel storage

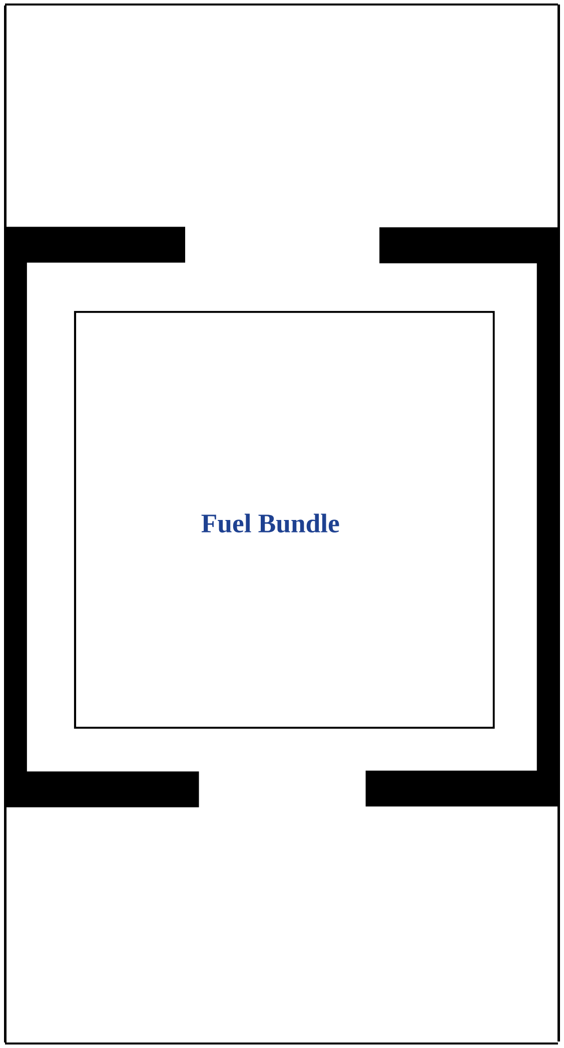
New Fuel Storage (LDFS)

- Aluminum or Stainless Steel
- Normal conditions
  - Dry
  - Centered fuel
- Numerous designs



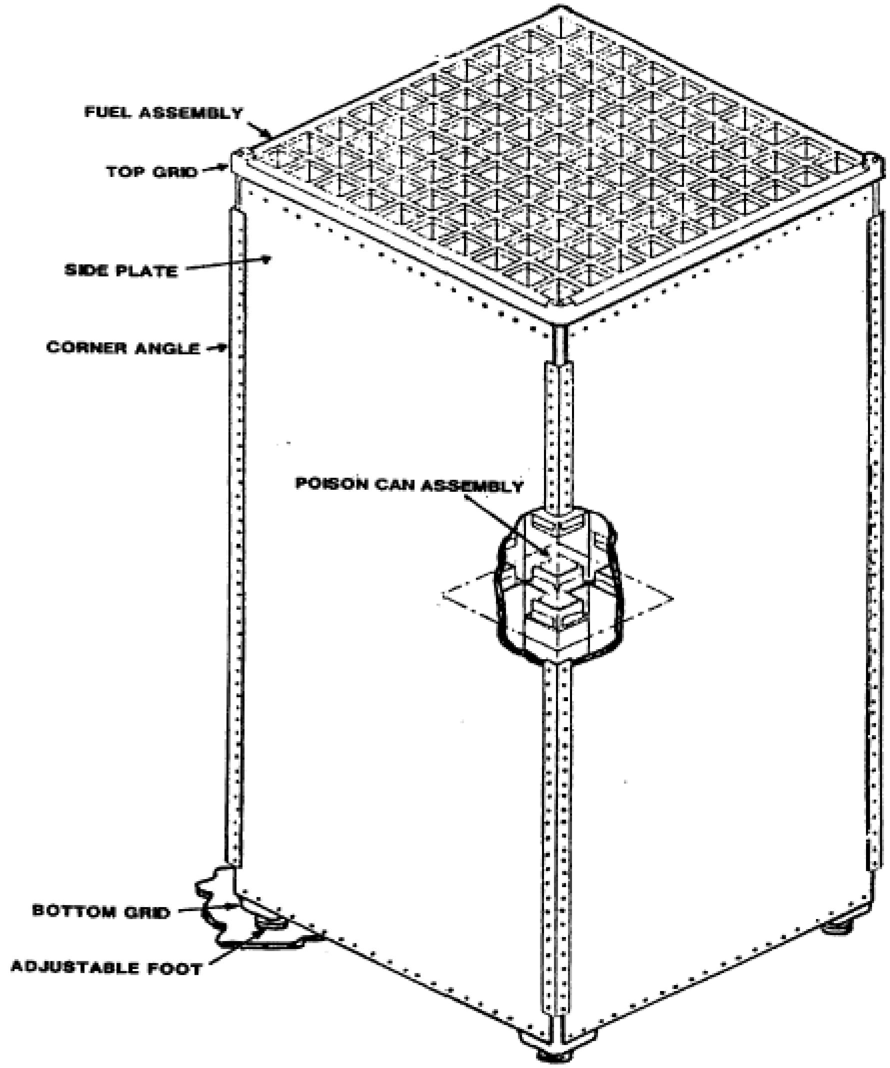
Containment Fuel Storage (LDFS)

- Aluminum or Stainless Steel
- Used for fresh and spent fuel
- Normal conditions
  - Wet
  - Centered fuel



Spent Fuel Storage (HDFS)

- Borated SS or SS with Boral or Boraflex
- Normal conditions
  - Wet
  - Centered fuel
- Numerous designs



Fuel Storage Objectives

- Cooling
- Shielding
- Preventing criticality accidents
- For BWR's; Establish the bundle design limit for :
  - New fuel storage
  - Core offload

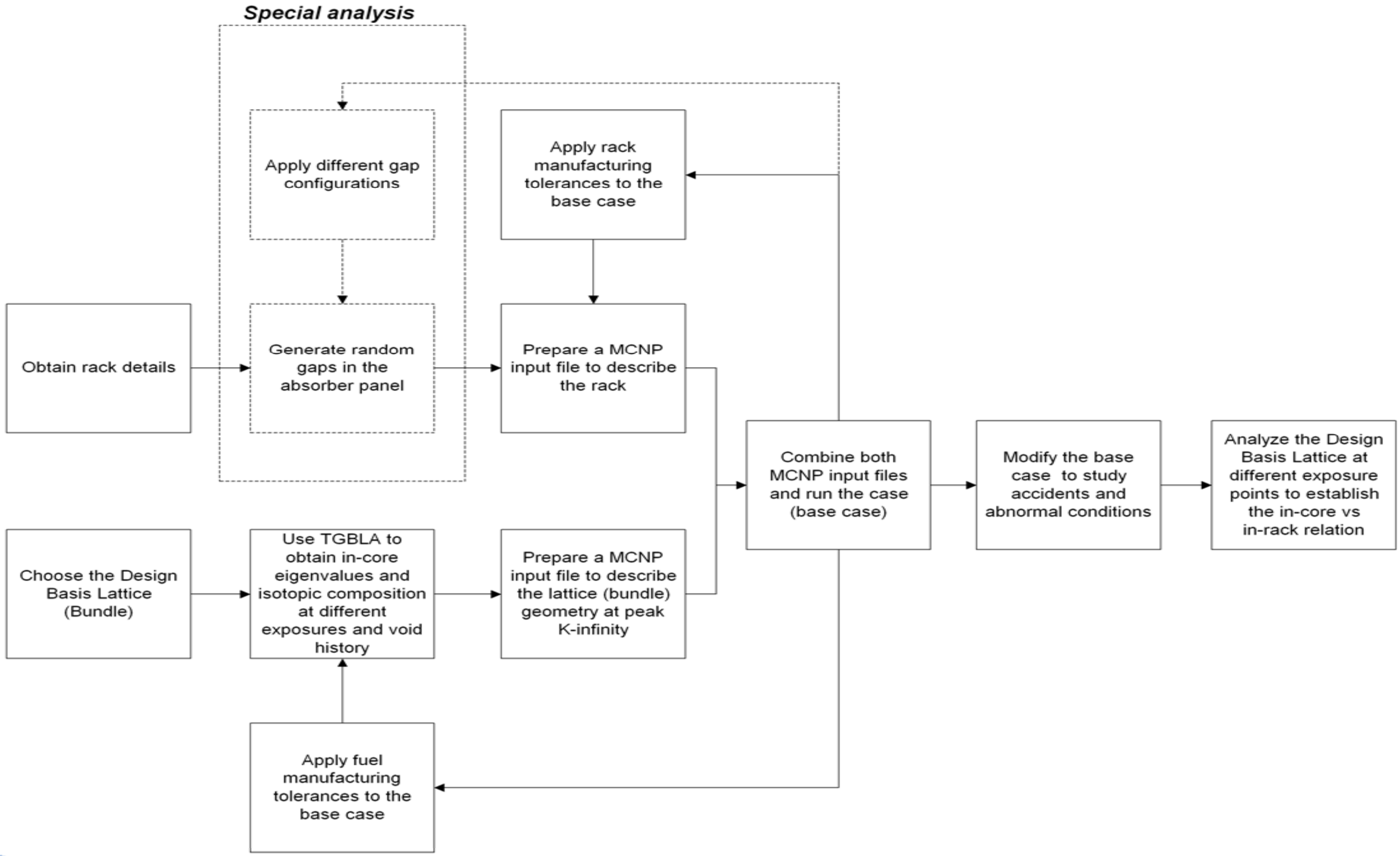
Global Nuclear Fuel
A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

TGBLA

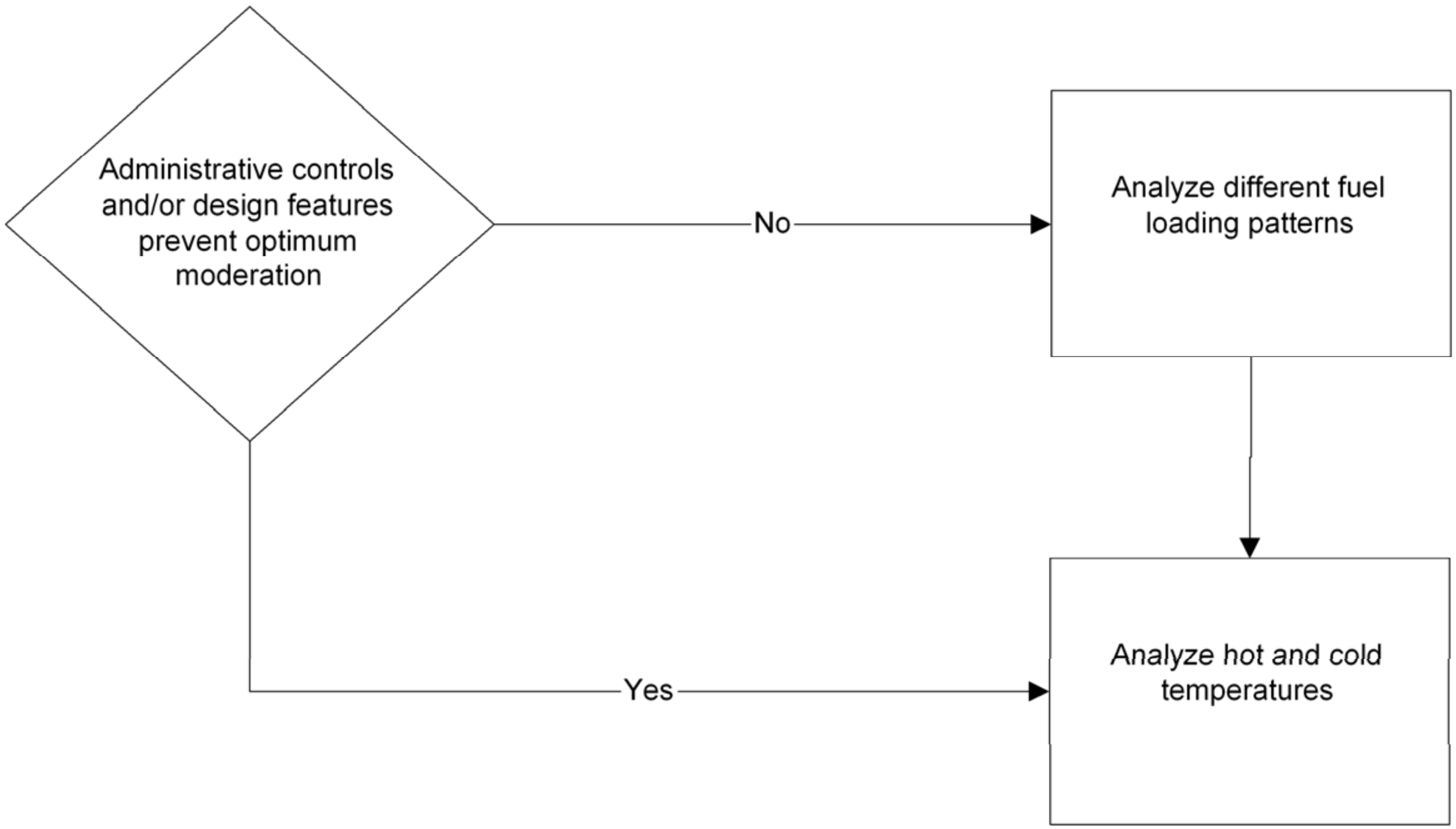
- GNF NRC approved depletion and lattice physics code
- Two-dimensional lattice design computer program for BWR fuel bundle analysis
- Output includes
  - Neutron Balance
  - Fission Density
  - Power Distribution
  - Exposure Distribution
  - Gamma Source
- Output used for GNF design, licensing, and core monitoring applications.

Overall Process



Additional Requirements for Dry Storage

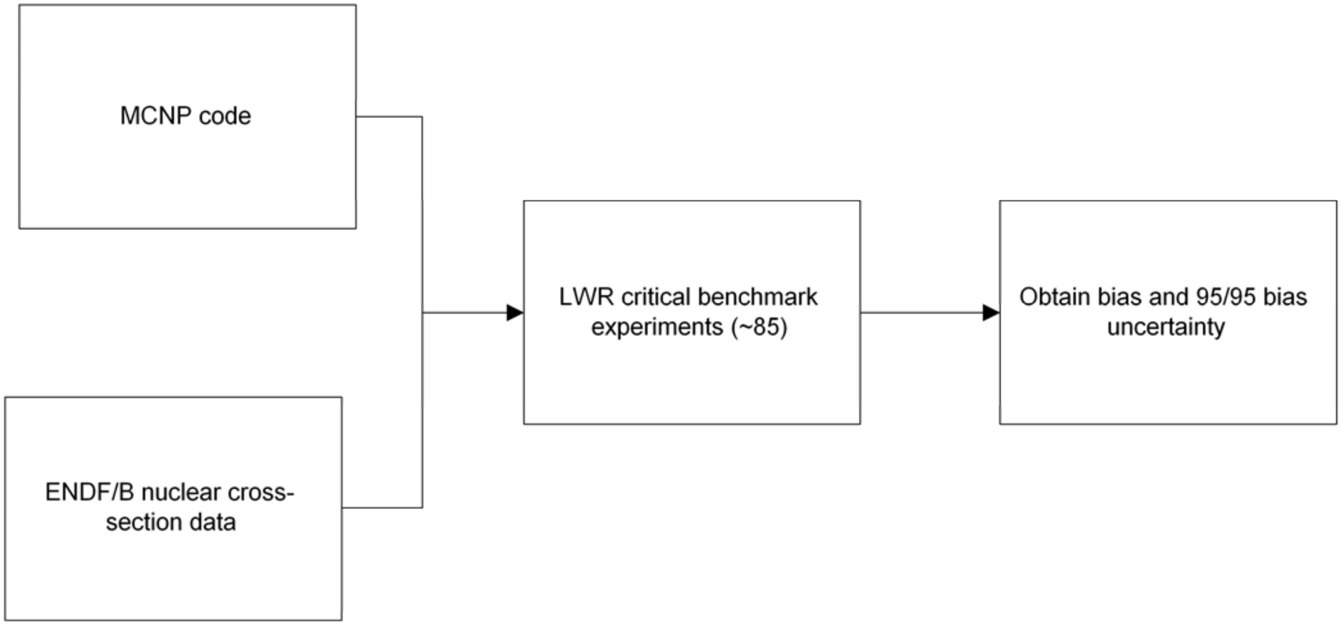
10 CFR 50.68



Global Nuclear Fuel
A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

Code/Data Bias



This process is repeated for every code/data combination

Benchmarks include:

- Borated steel plates
- Various enrichment and Gad content
- Cold and simulated hot
- Multi-lattice

Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

Absorber Sheet Degradation

- High radiation fields
- Water ingression
- Loss of boron and silica
- Panel shrinkage and Gaps

Blackness or BADGER Test

Obtain the probability distribution of:

- Number of gaps
- Gap size
- Gap location
- Areal density

Spent Fuel Storage Rack Modeling

- MCNP (2D or 3D)
- Rack structure
- Lattice structure
- Gap definition (or apply penalty)
- Bias
- Tolerances
- Uncertainties

Global Nuclear Fuel
A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

Criticality Analysis

Conservative assumptions

- Most reactive (lattice) bundle acceptable in storage rack
- No natural uranium
- No lumped fission products (TGBLA)
- No (or minor) structural material
- No neutron leakage (where applicable)
- Non-borated water
- Absorber density set to 95/95 minimum
- Panel dimensions set to minimum as-built

Criticality Analysis

$$K_{max(95/95)} = K_{mc} + \Delta K_{Bias} + \Delta K_{Tolerance} + \Delta K_{Uncertainty}$$

$$K_{max(95/95)} \leq 0.95$$

K_{max} - Maximum reactivity (95/95) in the rack;

K_{mc} - Eigenvalue from Monte Carlo calculation;

Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

Criticality Analysis

$$K_{max(95/95)} = K_{mc} + \Delta K_{Bias} + \Delta K_{Tolerance} + \Delta K_{Uncertainty}$$

$$\Delta K_{Bias} = \sum_{i=1}^{6} \Delta K_{Bi}$$

Δk_{B1} = Critical benchmark bias

Δk_{B2} = Depletion Credit

Accident or Abnormal Condition Bias

Δk_{B3} = Non-channeled assembly

Δk_{B4} = Moderator temperature variation

Δk_{B5} = Eccentric assembly location

Δk_{B6} = Horizontally dropped assembly

Δk_{B7} = Vertically dropped assembly

Δk_{B8} = Periphery placed assembly

Δk_{B9} = Aluminum rack box

Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

Criticality Analysis

$$K_{max(95/95)} = K_{mc} + \Delta K_{Bias} + \Delta K_{Tolerance} + \Delta K_{Uncertainty}$$

$$\Delta K_{Tolerances} = \sqrt{\sum_{i=1}^{7} \Delta K_{Ti}^{2}}$$

Δk_{T1} = Fuel enrichment
Δk_{T2} = Fuel pellet density
Δk_{T3} = Fuel pellet diameter
Δk_{T4} = Gadolinia content
Δk_{T5} = Clad thickness
Δk_{T6} = Rack wall thickness
Δk_{T7} = Rack pitch

$$\Delta K_{Uncertainty} = \sqrt{\Delta K_{U1}^{2} + \Delta K_{U2}^{2}}$$

Δk_{U1} = Critical benchmark bias uncertainty
Δk_{U2} = Problem uncertainty

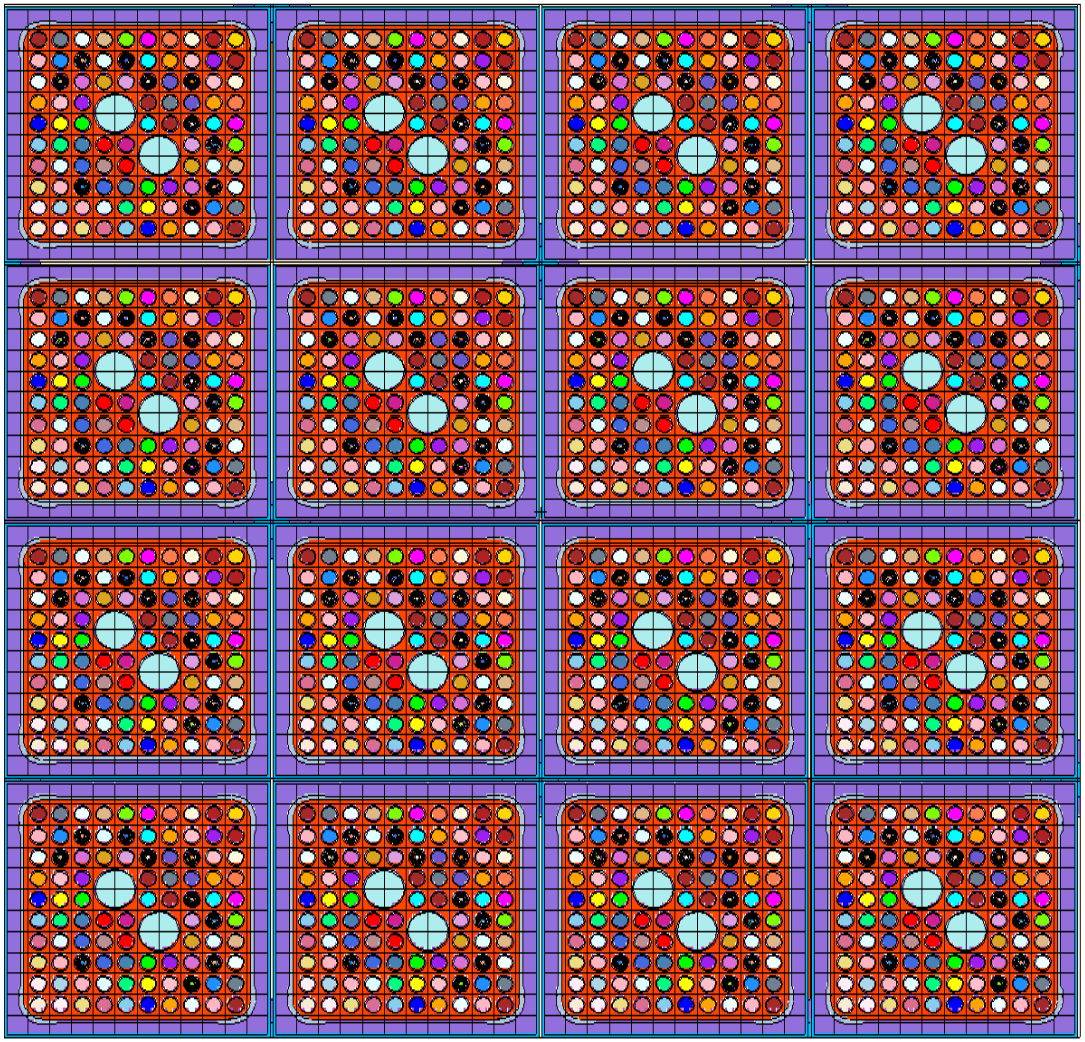
All tolerances and uncertainties shall be expressed at the 95/95 tolerance limit

Global Nuclear Fuel
A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

Normal Loading

Channels are modeled



Gap Modeling

Modeled gaps are shown in the circles

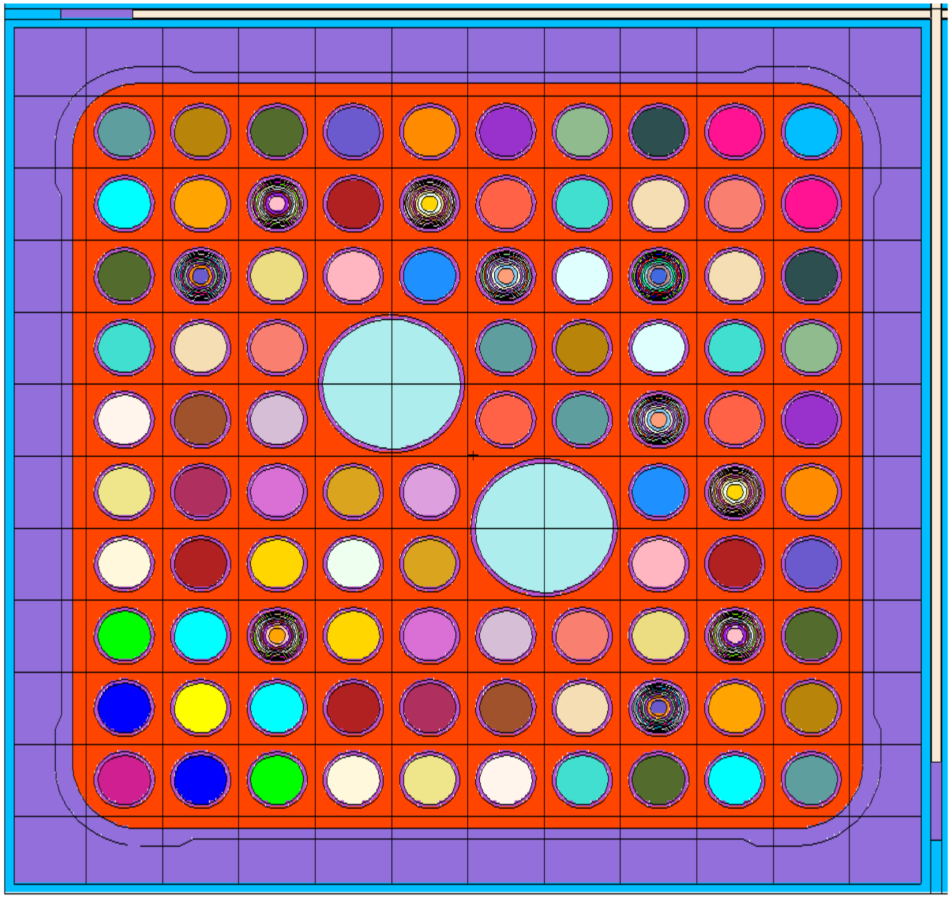
Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

Non-channeled

Channel material is replaced by water



Eccentric Loading

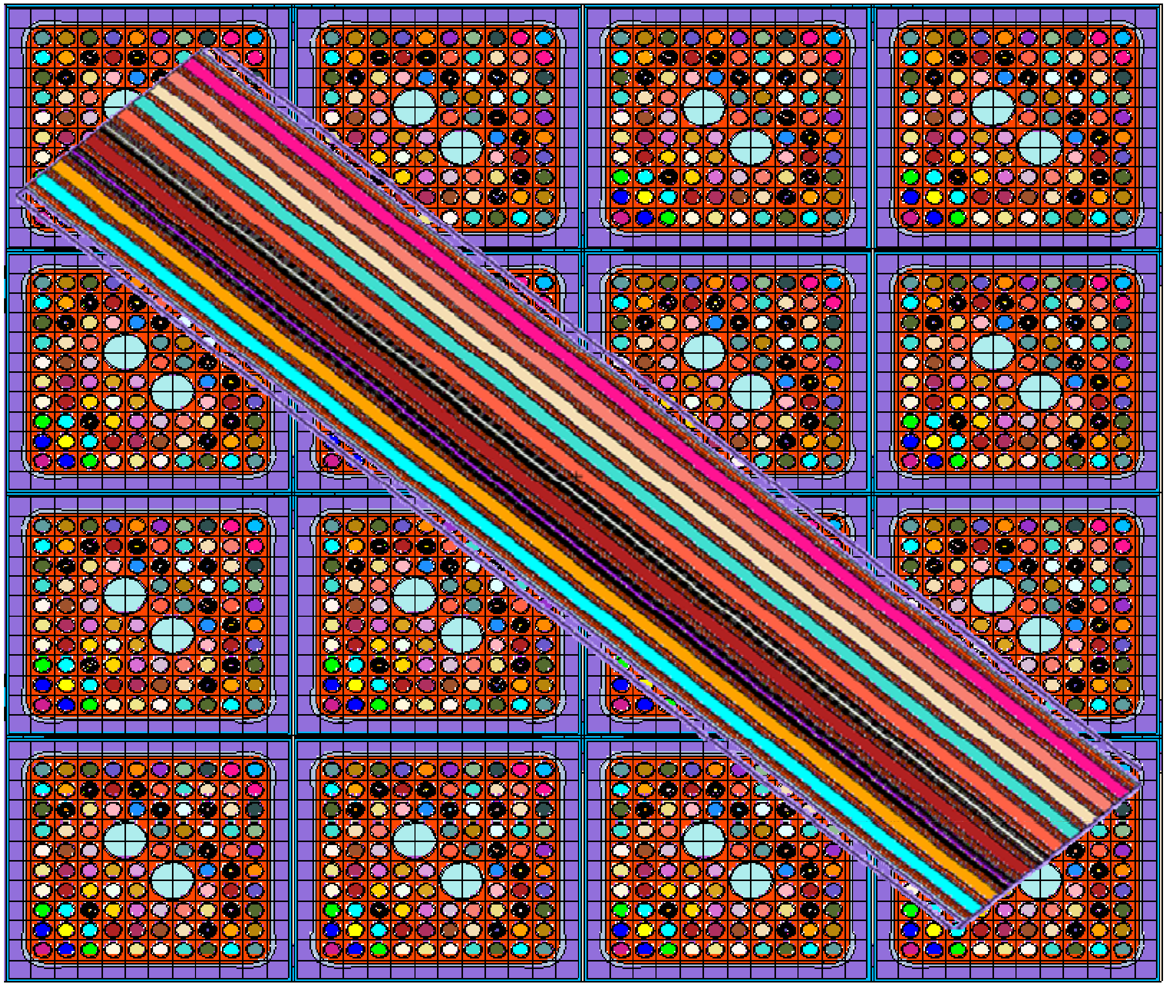
This is one scenario
for eccentric loading

Global Nuclear Fuel
A Joint Venture of GE, Toshiba, & Hitachi

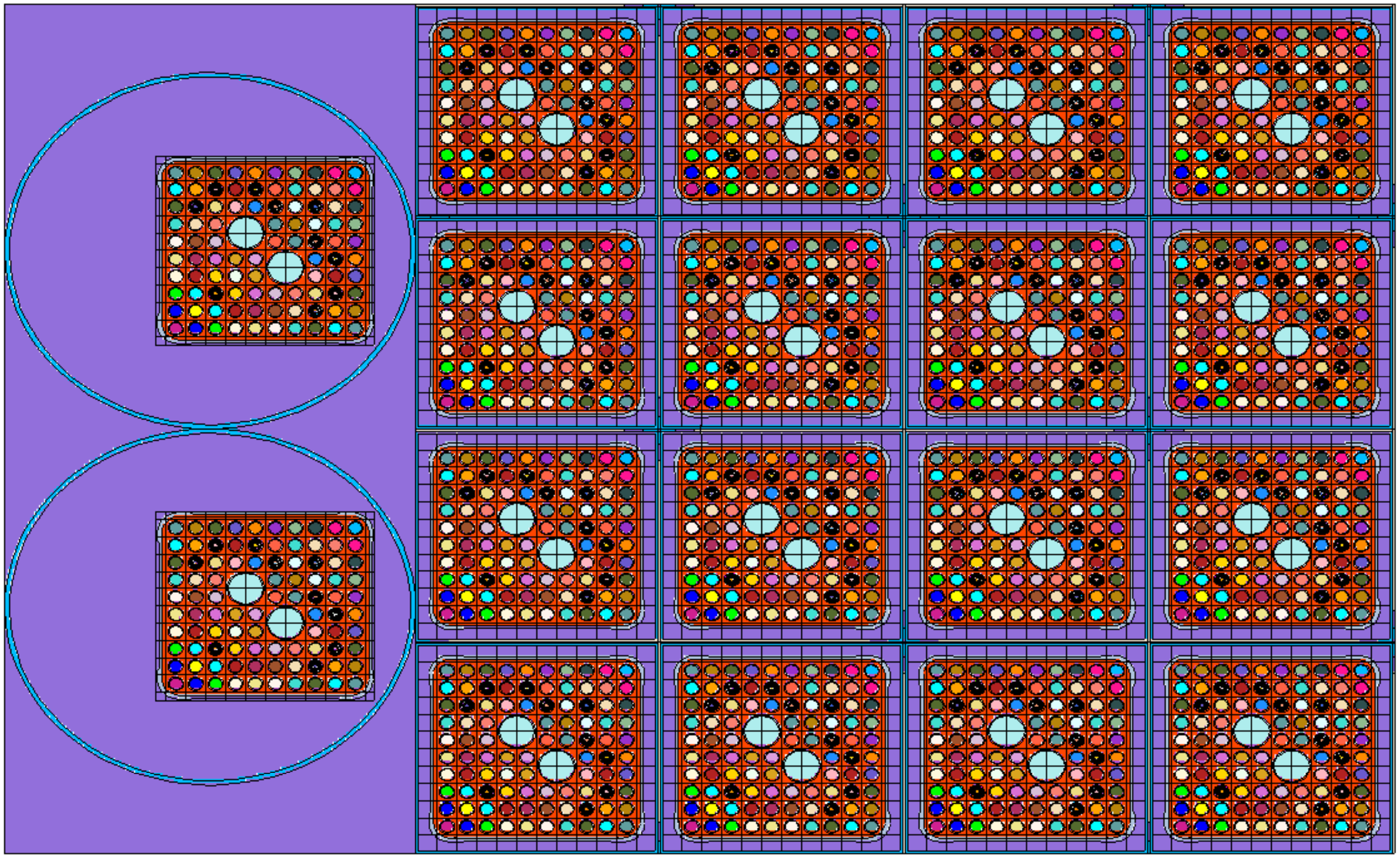
HITACHI

Dropped Bundle

The water separating the rack and the dropped bundle is not shown in the figure



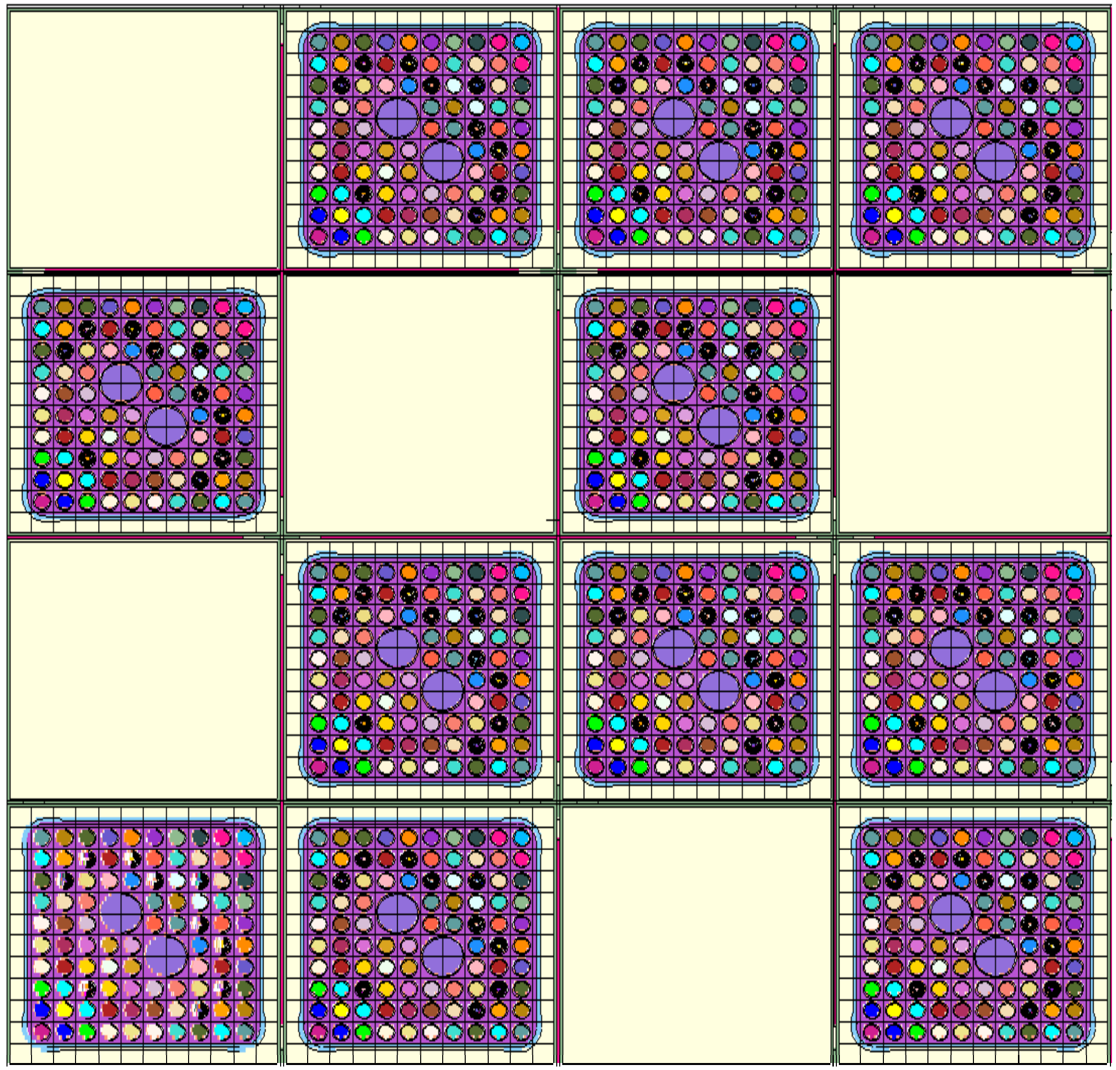
Abnormal Assembly Positioning



Bundles alongside the rack

Partial Loading

Applicable to LDFS racks under optimum moderation conditions and highly degraded HDFS racks



Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

Results

Case	Name	K	Δk	Effect
Base Case	K_{mc}	0.9241	0	--
Bias - Code				
Critical benchmark for MC code	Δk_{B1}			
Depletion credit	Δk_{B2}			+ ve
Total Bias - Misc.		=	0.005	
Bias - Abnormal Conditions				
Non-channeled assemblies	Δk_{B3}			- ve
Temperature increase to 100 ^oC	Δk_{B4}	0.9073	-0.0168	- ve
Temperature decrease to 4 ^oC	Δk_{B4}	0.9248	0.0007	+ ve
Eccentric loading	Δk_{B5}			- ve
Total Bias - Abnormal		=	0.0007	
Bias - Accident Conditions				
Horizontally dropped bundle	Δk_{B6}			+ ve
Vertically dropped bundle	Δk_{B7}			+ ve
Periphery dropped assembly (Near)	Δk_{B8}			- ve
Periphery dropped assembly (Far)	Δk_{B8}			- ve
Total Bias - Accident		=	0.007	

Tolerances				
Fuel enrichment increase	Δk_{T1}			+ ve
Fuel pellet density increase	Δk_{T2}			+ ve
Gadolinina wt% decrease	Δk_{T3}			+ ve
Clad thickness increase	Δk_{T4}			- ve
Clad thickness decrease	Δk_{T4}			+ ve
Rack wall increase	Δk_{T5}			- ve
Rack pitch decrease	Δk_{T6}			+ ve
Total Tolerances		=	0.009	
Uncertainties				
Critical benchmark bias for MC code	Δk_{U1}	--		
Problem Specific Error	Δk_{U2}	--		
Total Uncertainty		=	0.002	
K_{max}		=	**0.9478**	

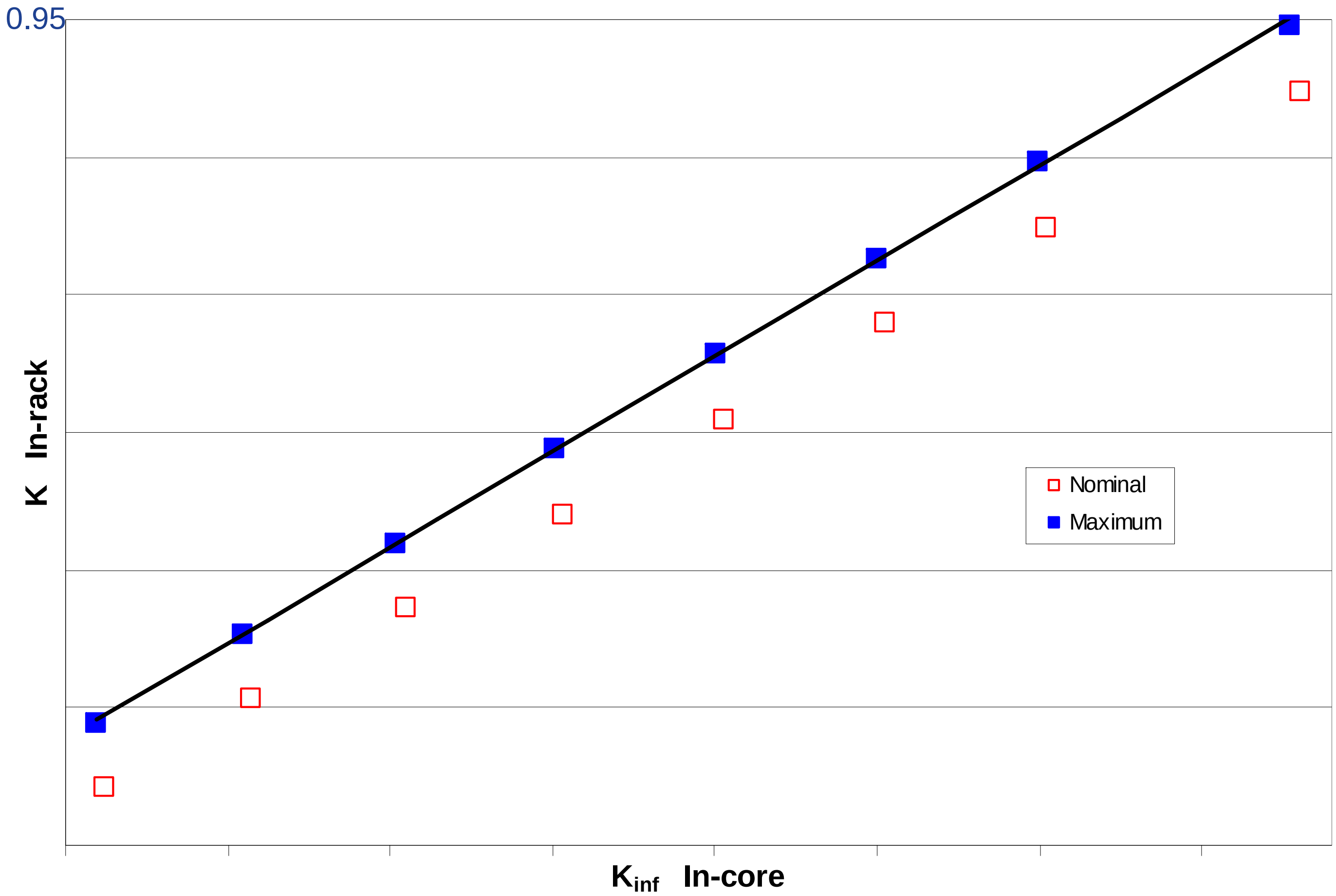
*** Negative effects (relative to the base case) are not included in the rollup of ΔK.**

Global Nuclear Fuel

A Joint Venture of GE, Toshiba, & Hitachi

HITACHI

Results



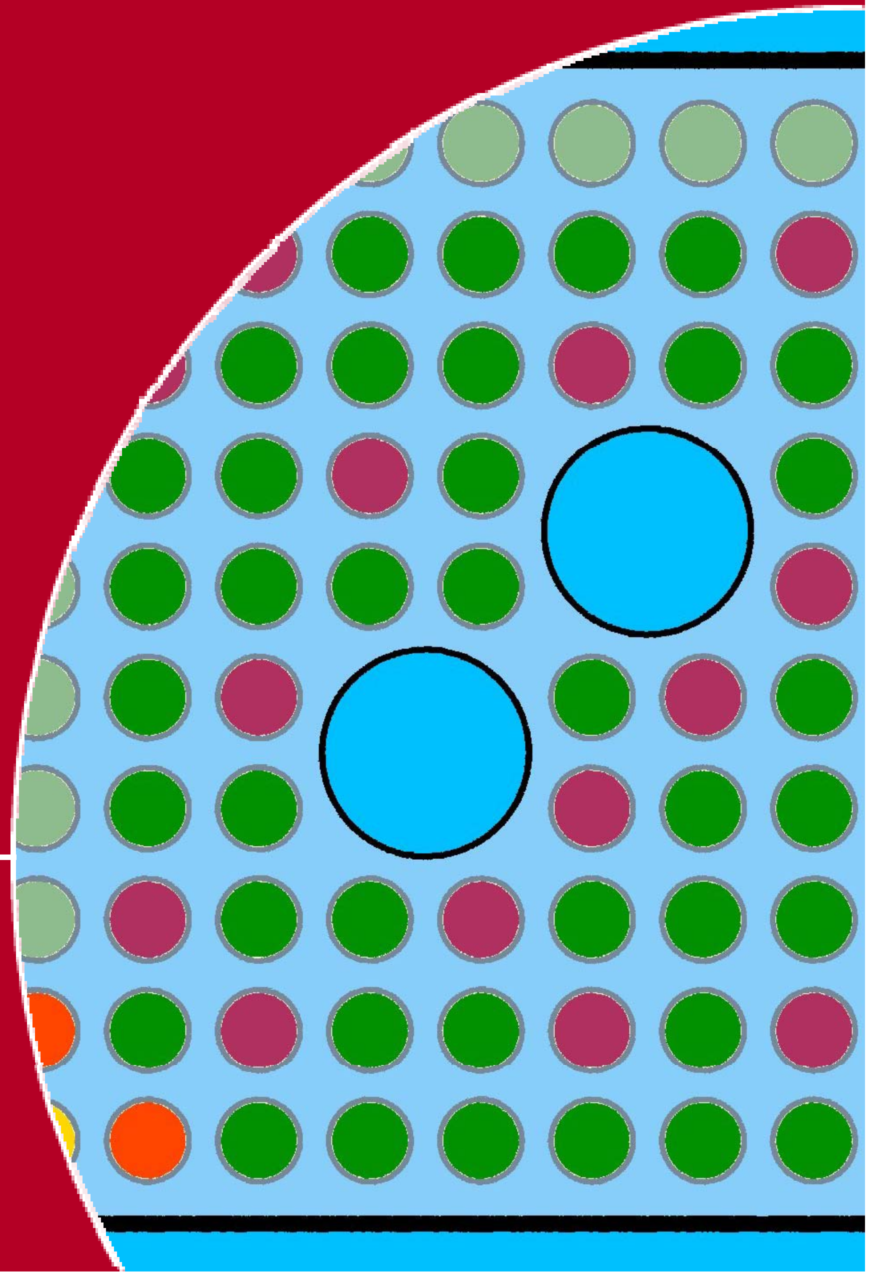
Conclusions

- TGBLA is a NRC approved depletion and lattice physics code.
- The most reactive (lattice) bundle acceptable in storage rack is used in the criticality analysis.
- Two or three-dimensional MCNP models are used to evaluate the storage racks.
- Sub-criticality must be ensured at all times in storage racks.
- All credible scenarios are taken into consideration.

Questions

Studsvik

CASMO: Studsvik's Lattice Physics Code



Dr. Kord S. Smith
NEI/NRC Presentation
May 1, 2009

Overview

- History of CASMO
- CASMO applications
- Examples of sensitivity to code versions/libraries
- CASMO benchmarking:
  - BOL PWR criticals
  - BOL BWR criticals
  - BOL storage rack criticals
  - MCNP/ORIGEN depletion comparisons
  - Measured isotopics comparisons
- Examples of depletion sensitivity to codes/libraries
- In-core reactor benchmarking
- Summary

CASMO Customer Base

AREVA

Siempelkamp
NIS Ingenieurgesellschaft mbH

NuScale Power

OPPD
Omaha Public Power District

APS

okg
- a company in the E.ON-group

FirstEnergy

TÜV NORD

Dominion
It all starts here.

DTE Energy

The Japan Atomic Power Company

INL Idaho National Laboratory

TOKYO ELECTRIC POWER COMPANY
TEPCO

IBERDROLA

MITSUBISHI HEAVY INDUSTRIES, LTD.

HOLTEC INTERNATIONAL

THE KANSAI ELECTRIC POWER CO.,INC.

Progress Energy

Exelon.

SOUTHERN COMPANY

NOK

KYUSHU ELECTRIC POWER CO.,INC.

VATTENFALL

mvm paks nuclear power plant

Duke Energy

FPL.

Xcel Energy
RESPONSIBLE BY NATURE

e·on

Nebraska Public Power District
Always there when you need us

National Nuclear Laboratory

TVO

CFE Comisión Federal de Electricidad

Institute of Nuclear Energy Research
INER

Entergy

TÜV SÜD

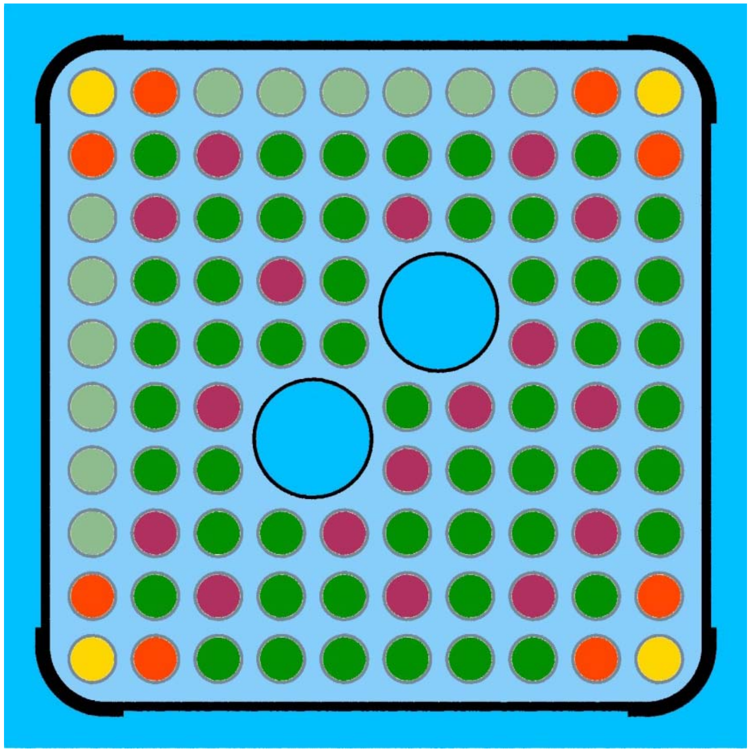
Constellation Energy

SOUTHERN CALIFORNIA EDISON
An EDISON INTERNATIONAL Company

RWE The energy to lead

CASMO:

<u>simple inputs</u>, automated deletion, SFP rack branches, deterministic solutions



BWR Bundle (10x10, 7 wt% Gd)

```
TTL TFU=920.7 TMO=561.5 VOI=40 * GE14-EXAMPLE
BWR 10 1.3 13.4 0.19 0.71 0.72 1.33/0.3048 3.8928
PIN 1  0.46 0.647 0.51
PIN 2  1. 1.24/'MOD' 'BOX'//4
LPI 1
  1  1
  1  1  1
  1  1  1  1
  1  1  1  1  1
  1  1  1  2  2  1
  1  1  1  2  2  1  1
  1  1  1  1  1  1  1  1
  1  1  1  1  1  1  1  1  1
  1  1  1  1  1  1  1  1  1  1
FUE 1  10.5/2.40
FUE 2  10.5/3.60
FUE 3  10.5/4.40
FUE 4  10.5/4.90
FUE 5  10.2/4.90  64016=7.0
LFU 1
  2  4
  3  5  4
  3  4  4  5
  3  4  4  4  4
  3  4  5  0  0  4
  3  4  4  0  0  5  4
  3  5  4  4  5  4  4  4
  2  4  5  4  4  5  4  5  4
  1  2  4  4  4  4  4  4  2  1
CRD 0.41 0 1.98 10.4 0.21 0.57/'B4C' 'CRS'//'CRD' 'ROD'
DEP -70.
END
```

CASMO is a “package” of many calculational models

Input Processor

↓

Neutronics Data Library (NJOY)

↓

Resonance Calculations

↓

Pin Cell Calculations

↓

2-D Transport Calc. (MoC)

↓

Isotopic Depletion/Decay

↓

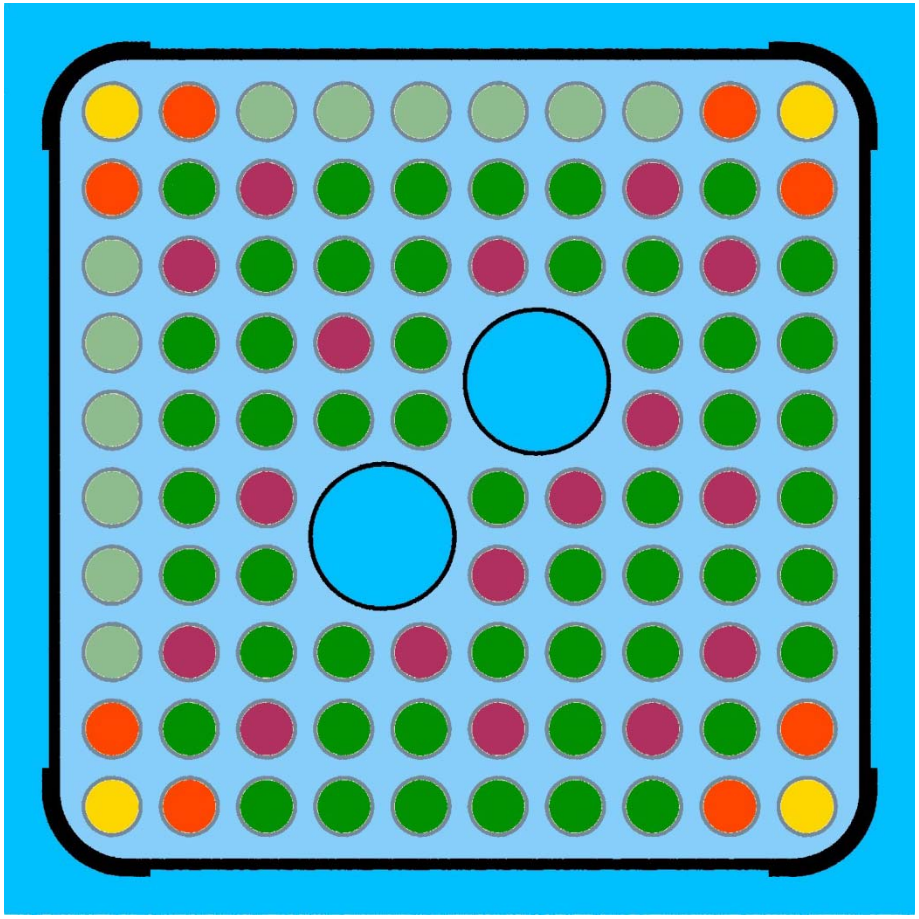
Detailed Edits

↓

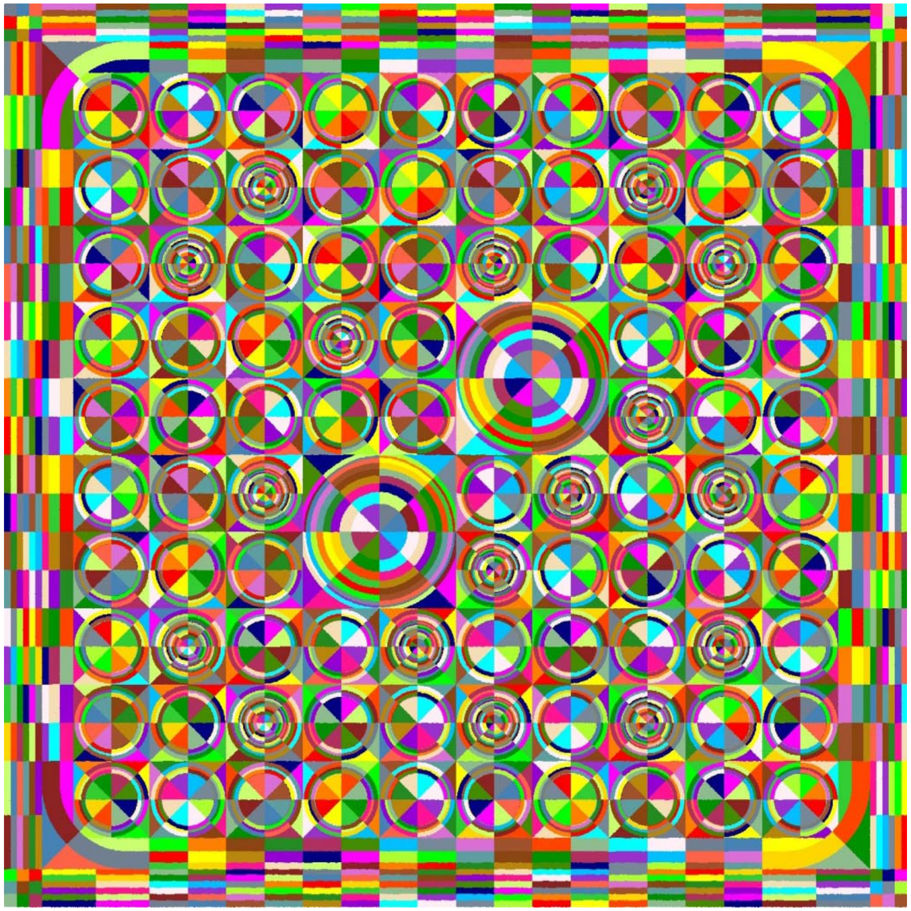
Restart Database for Branches

CASMO: Lots of Geometric Detail

Material Regions



Flat Source Regions



CASMO has been used for more than 30 years

CASMO circa 1978 ➔ first in-house applications at Studsvik, Sweden

CASMO-2 circa 1981 ➔ 25 group library (**ENDF/B-III**), 2-D transport: transmission probability, homogeneous geometry, external Gd depletion, Fortran-IV

CASMO-3 circa 1985 ➔ 40 group library (**ENDF/B-IV**), 2-D transport: transmission probability, homogeneous geometry, external Gd depletion, **2x2 bundle capability,** data for **SIMULATE-3**, F66

CASMO-4 circa 1993 ➔ 70 group library (ENDF/B-IV), **2-D transport: MoC**, **heterogeneous geometry, internal Gd depletion**, F77

CASMO-4E circa 2001 ➔ 70 group library (ENDF/B-IV,**ENDF/B-VI,JEF2**), 2-D transport: MoC, heterogeneous geometry, internal Gd depletion, **MxN general multi-assembly, Pn-scattering**, F90

CASMO-5 circa 2007 ➔ **586 group library (ENDF/B-VII**), 2-D transport: MoC, heterogeneous geometry, internal Gd depletion, multi-group data for **SIMULATE-5**, MxN multi-assembly, Pn-scattering, **Spent Nuclear Fuel edits, Fortran-95**

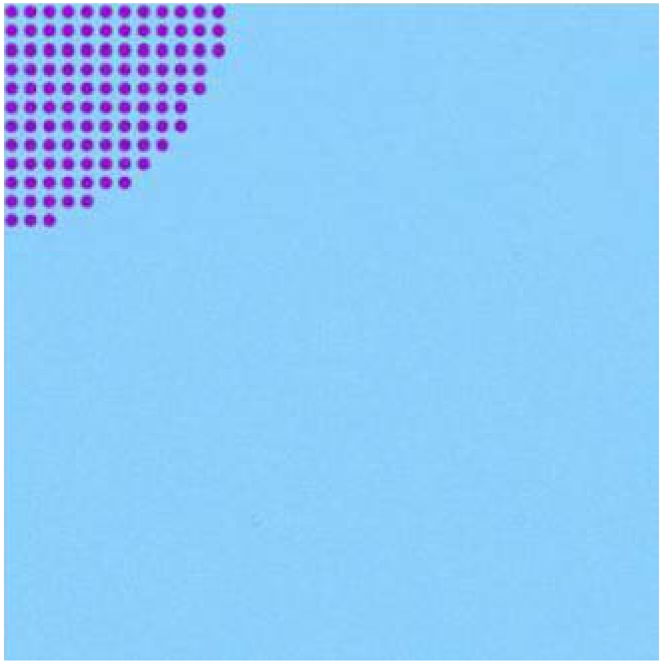
CASMO 17x17 PWR Sensitivity Calculations

	(Mwd/kg)	C3/E4	C4/E4	C4/E6A	C4/E6U	C4/J2	C5/E7
Bor=500	0.1	1.33186	1.33126	1.33217	1.3266	1.33563	1.33346
	40	0.99961	1.00021	0.99435	0.99312	0.99766	0.99878
	80	0.78065	0.78633	0.7794	0.78368	0.78559	0.78936
plus spacer	0.1	1.32474	1.32412	1.32448	1.31891	1.32789	1.32565
	40	0.99486	0.99545	0.9894	0.98821	0.9927	0.99389
	80	0.77844	0.78415	0.77753	0.78181	0.7837	0.78759
Spacer worth (pcm)	0.1	-404	-405	-436	-440	-436	-442
	40	-478	-478	-503	-500	-501	-493
	80	-364	-354	-309	-305	-307	-285
BOR=2000	0.1	1.20067	1.19931	1.19946	1.19451	1.20298	1.19853
	40	0.91642	0.91677	0.91183	0.91169	0.91574	0.91521
	80	0.73248	0.73813	0.7323	0.73698	0.73864	0.74146
plus spacer	0.1	1.19577	1.19441	1.19411	1.18916	1.19758	1.19312
	40	0.91353	0.91389	0.90886	0.90873	0.91274	0.91232
	80	0.7315	0.73716	0.73167	0.73632	0.73796	0.74092
Spacer worth (pcm)	0.1	-341	-342	-374	-377	-375	-378
	40	-345	-344	-358	-357	-359	-346
	80	-183	-178	-118	-122	-125	-98
Boron worth (pcm)	0.1	8204	8264	8305	8336	8256	8443
	40	9081	9100	9101	8994	8967	9142
	80	8424	8304	8252	8086	8091	8184

- **Reactivity differences are insensitive to code versions, energy group structures, and nuclear data libraries.**

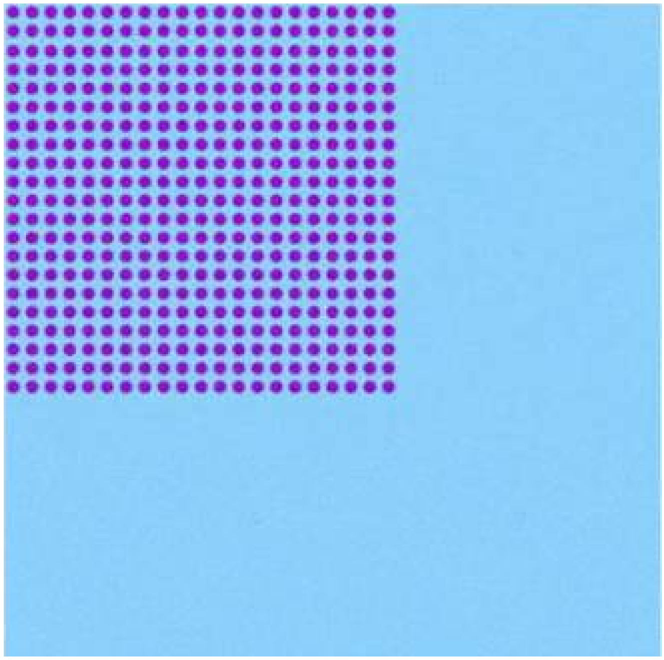
Benchmarking: B&W Simple Pin Cell Criticals

Core I, Boron = 0



K-eff=0.99913
(35% radial leakage)

Core II, Boron = 1037

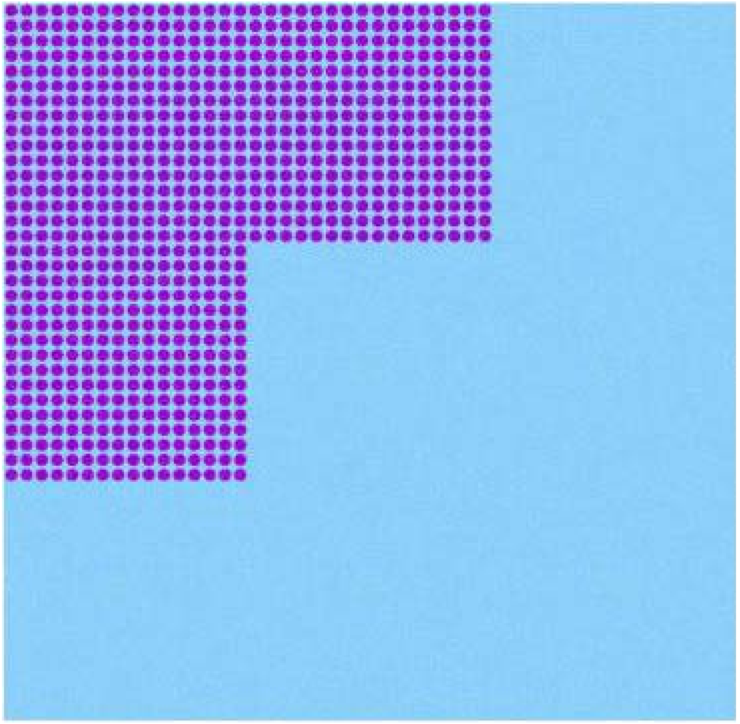


K-eff=1.00059
(15% radial leakage)

Radial leakage is well predicted.

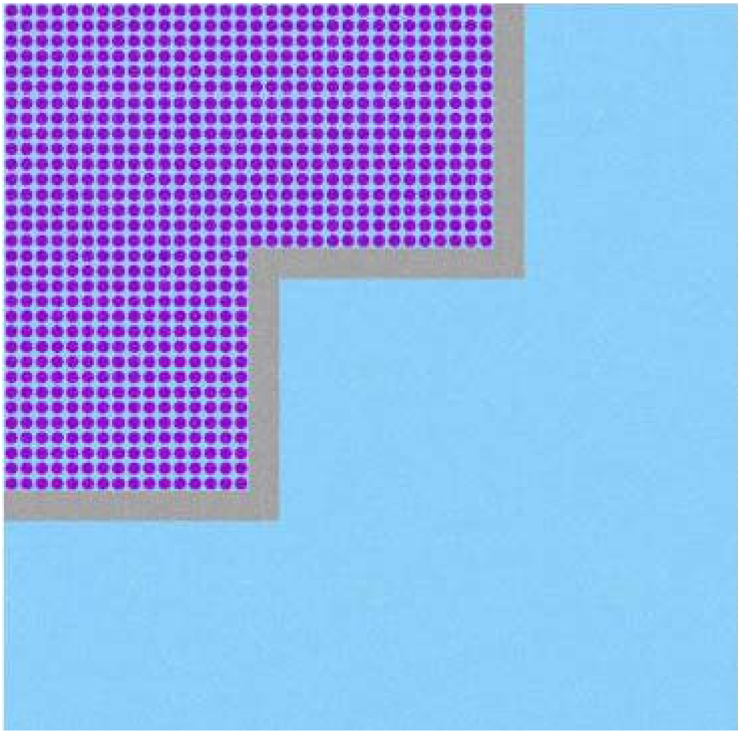
Dimple Baffle/Reflector Criticals

Core S06a (No Baffle)



K-eff=1.00125

Core S06B (Baffle)

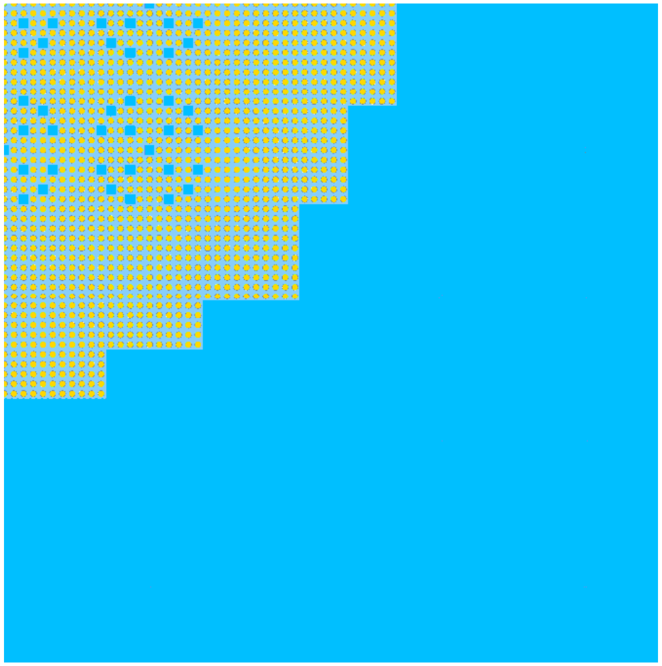


K-eff=1.00058

Baffle/reflector effects are well predicted.

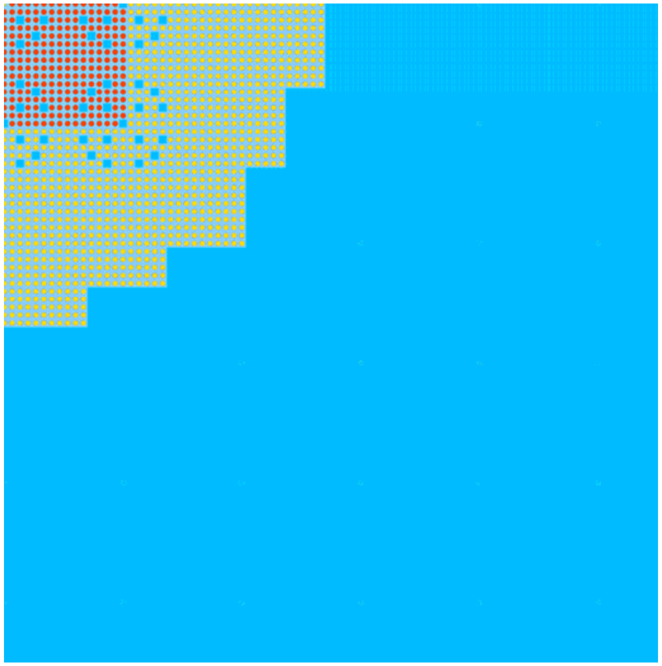
B&W 1810 Heterogeneous Criticals

Core 0I



Single Region

Core 12



Two Region

Excellent tests of BOL cold fuel assembly reactivity.

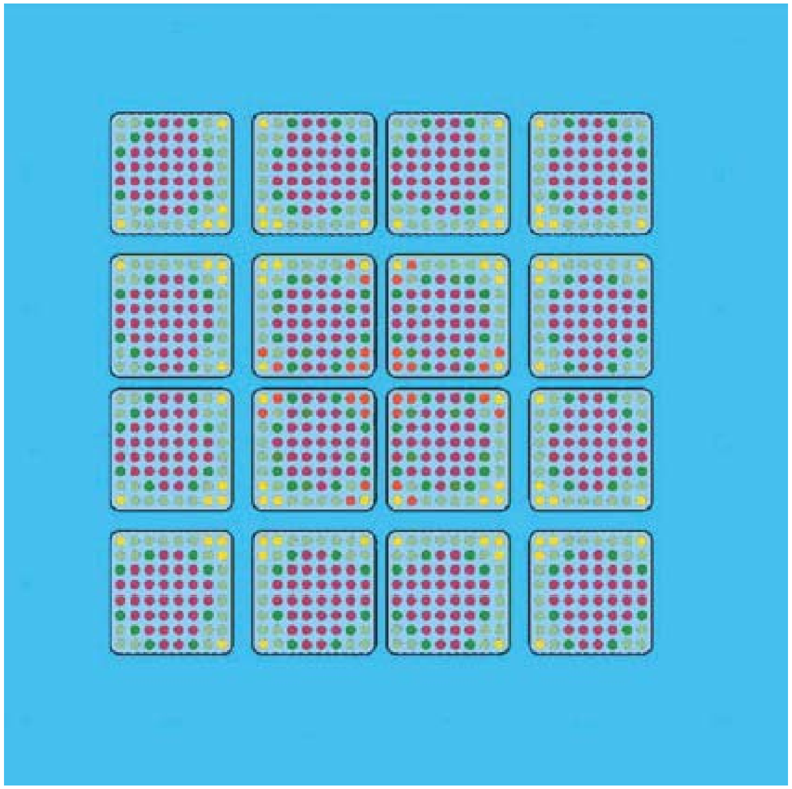
Summary of B&W 1810 Criticals

Core	Boron (PPM)	# 4% Gd Pins	# of AIC Rods	CASMO k-eff	Fission Rate Total RMS (%)
01	1337.9	--	--	1.00083	0.51
02	1250.0	--	16	1.00027	
03	1239.3	20	--	1.00047	
04	1171.7	20	16	1.00106	
05	1208.0	28	--	1.00018	
05A	1191.3	32	--	1.00008	0.57
05B	1207.1	28	--	1.00025	
06	1155.8	28	16	1.00037	
06A	1135.6	32	16	1.00031	
07	1208.8	28	--	1.00019	
08	1170.7	36	--	1.00028	
09	1130.5	36	16	1.00015	
10	1177.1	36	16	1.00010	
12	1899.3	--	--	1.00114	0.69
13	1635.4	--	16	1.00156	
14	1653.8	28	16	1.00084	0.79
15	1479.7	28	16	1.00140	
16	1579.4	36	---	1.00081	
17	1432.1	36	16	1.00098	
18	1776.8	--	--	1.00268	0.86
19	1628.3	16	--	1.00235	
20	1499.0	32	--	1.00214	
Average (Cores 01-17)				1.00059	
Standard Dev. (Cores 01-17)				0.00047	
Average (Cores 18-20)				1.00239	
Standard Dev. (Cores 18-20)				0.00027	
Average (All Cores)				1.00084	
Standard Dev (All Cores)				0.00077	

Gadolinia, AIC rods, boron are well predicted.

KRITZ-4 - Real BWR Bundle Criticals

- Gap orientations
- Fuel enrichment
- Gad loadings
- Control rod in center
- Temperature

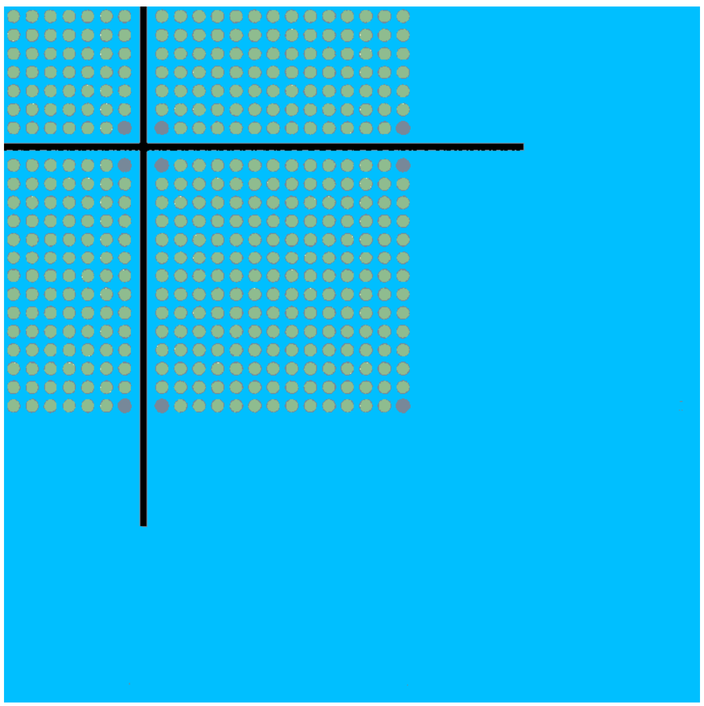


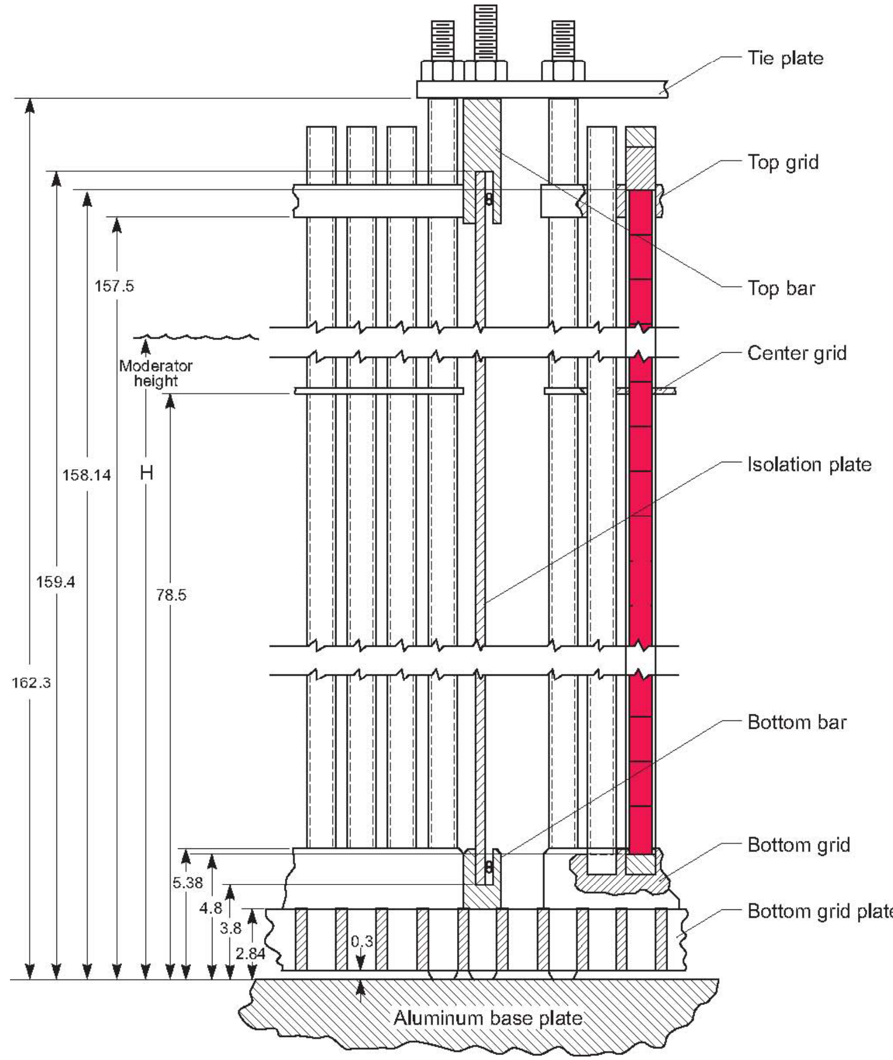
Kritz-4 BWR Criticals	
Ave k-eff (Cold) (31 cores)	0.99966
S.D. (Cold ~20 C)	0.00069
Ave k-eff (Warm) (11 cores)	0.99893
S.D. (Warm 80-100 C)	0.00056
Ave. k-eff (Hot) (17 cores)	0.99835
S.D. (Hot ~240 C)	0.00042
Ave. k-eff (All cores)	0.99915
S.D. (All Cores)	0.00083

BWR bundle reactivity, rods, temperatures are well predicted.

B&W 1484 - Storage Rack Criticals

- 1979, various configurations:
  - Moderator height
  - Bundle separation
  - Steel isolation sheets
  - Boral Plates





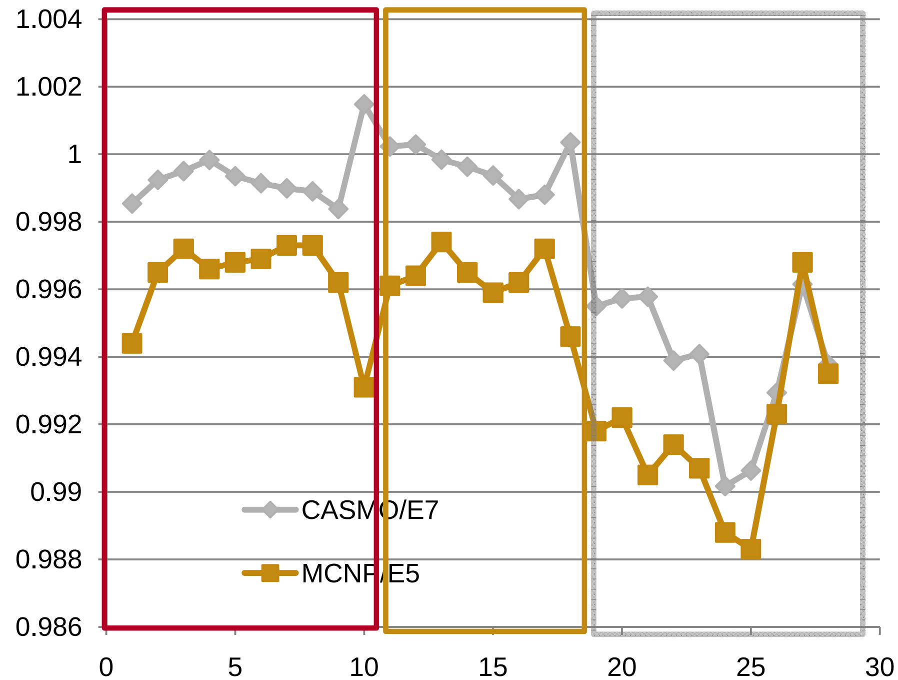
3-D MCNP-5 (ENDV/B-VII) vs. 2-D CASMO (B&W 1484 Boron/Water Height)

Core	Height	Boron	MCNP-5	s.d.	CASMO-5	C5-MCNP
	cm	ppm	K-eff	1 sigma	K-eff	K-eff
III-A	148.63	769	1.00062	0.00009	0.99965	-0.00097
III-B	144.88	764	1.00116	0.00010	0.99997	-0.00119
III-C	140.38	762	1.00100	0.00010	0.99951	-0.00149
III-D	131.32	753	1.00080	0.00010	0.99931	-0.00149
III-E	120.64	739	1.00115	0.00009	0.99918	-0.00197
III-F	110.04	721	1.00121	0.00010	0.99911	-0.00210
III-G	100.32	702	1.00073	0.00009	0.99863	-0.00210

2-D model of axial leakage using <u>geometrical bucking plus 11.0 cm</u> extrapolation length is an adequate substitute for measured bucklings.

B&W 1484 – MCNP/E5 and CASMO/E7

Core	CASMO/E7	MCNP/E5	Isolation Sheet
I	0.99854	0.9944	------
II	0.99924	0.9965	------
IIIA	0.9995	0.9972	------
IIIB	0.99983	0.9966	------
IIIC	0.99935	0.9968	------
IIID	0.99914	0.9969	------
IIIE	0.99899	0.9973	------
IIIF	0.9989	0.9973	------
IIIG	0.99838	0.9962	------
X	1.00148	0.9931	------
XIA	1.00023	0.9961	S.S.
XIB	1.00029	0.9964	S.S.
XIC	0.99984	0.9974	S.S.
XID	0.99963	0.9965	S.S.
XIE	0.99937	0.9959	S.S.
XIF	0.99867	0.9962	S.S.
XIG	0.9988	0.9972	S.S.
XII	1.00035	0.9946	S.S.
XIX	0.9955	0.9918	Al-1
XX	0.99573	0.9922	Al-1
XXI	0.99578	0.9905	Al-1
XVII	0.99389	0.9914	Al-2
XVIII	0.99408	0.9907	Al-2
XV	0.99017	0.9888	Al-3
XVI	0.99063	0.9883	Al-3
XIV	0.99294	0.9923	Al-4
XIII	0.99615	0.9968	Al-5
XIIIA	0.99377	0.9935	Al-5A



Boral sheet boron content:

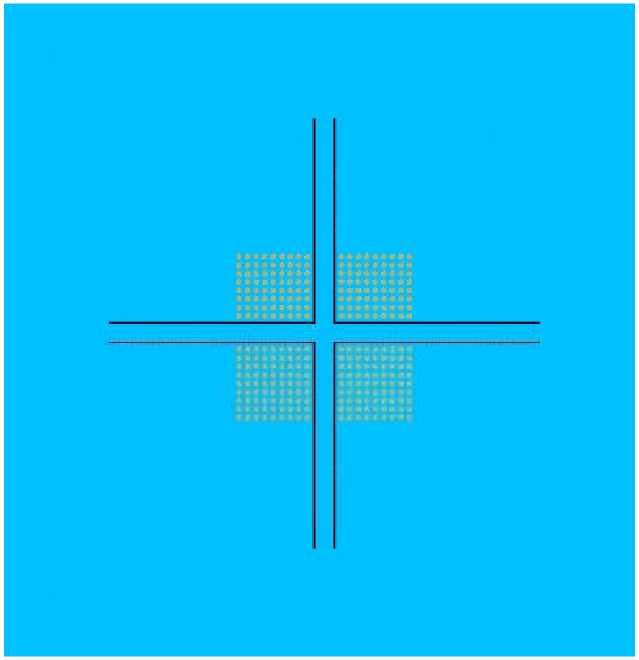
set 1 = 2.1% mean, 15% sd
set 2 = 1.4% mean, 8% sd
set 3 = 4.0% mean, 29% sd
set 4 = 1.2% mean, 6% sd
set 5 = 1.3% mean, 7% sd
set 5a = 1.4% mean, 8% sd

2-D CASMO accuracy is very similar to 3-D MCNP
(Note ENDF/B-V has -400 pcm bias relative to ENDF/B-VII)

PNL 6205 – Flux Trap/Rack Criticals

- **1988, Various configurations:**
  - **Bundle separation**
  - **Boral plate boron content**
  - **Extrapolated to critical**

Plate Separation	Boral gB/cm2	CASMO k-eff
0	0.05	1.00197
0	0.13	0.99886
3	0.13	0.99955
0	0.45	1.00270
3	0.45	0.99864

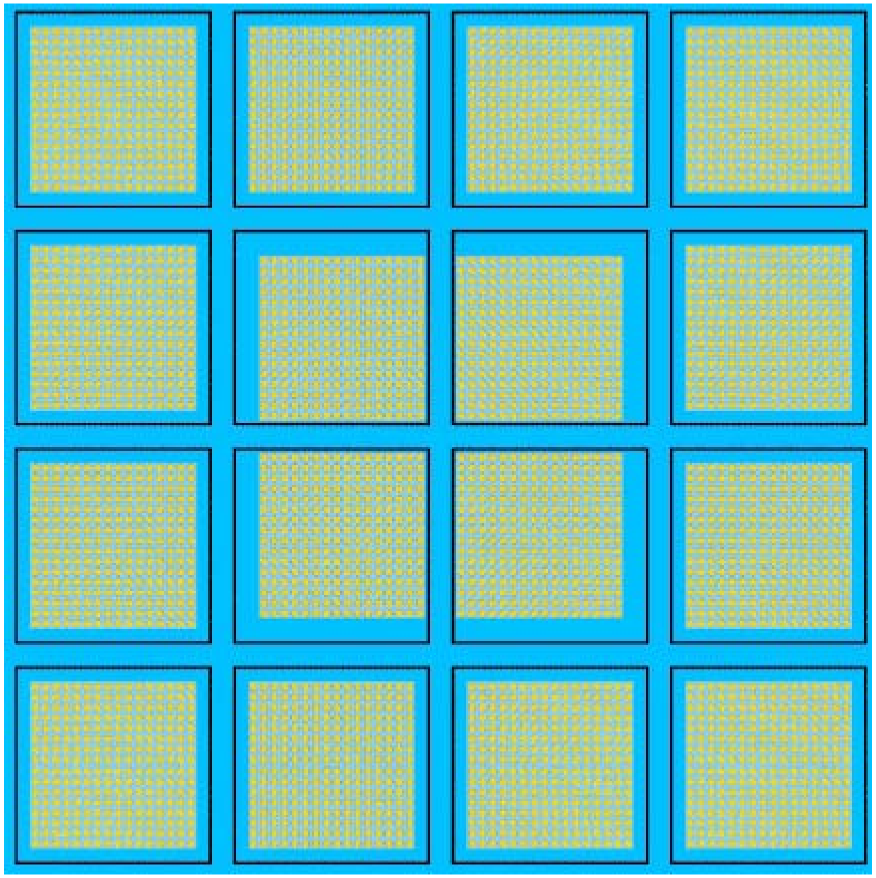


Flux trap geometries are predicted well with CASMO.

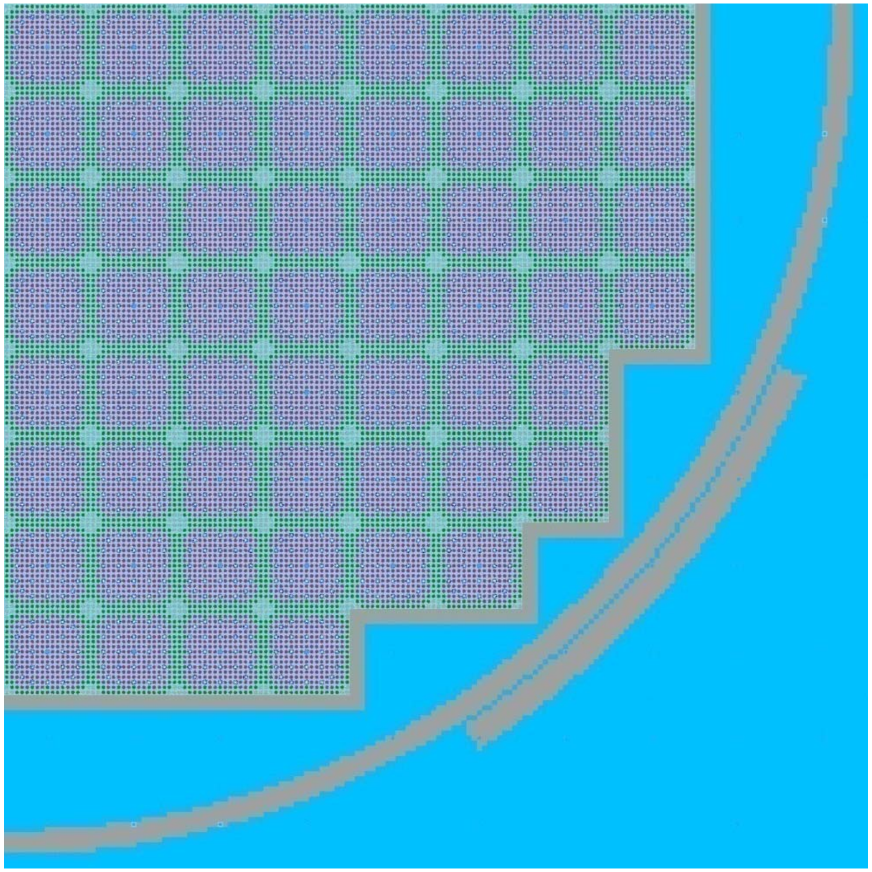
CASMO Multi-assembly Rack Models

- 2-D with or without axial bucking
- Any size regular rectangular rack
- Any number of rack material layers
- Arbitrary positioning of fuel bundles
- Fresh and/or depleted fuel
  - single-assembly CASMO depletion
  - MxN CASMO core depletion

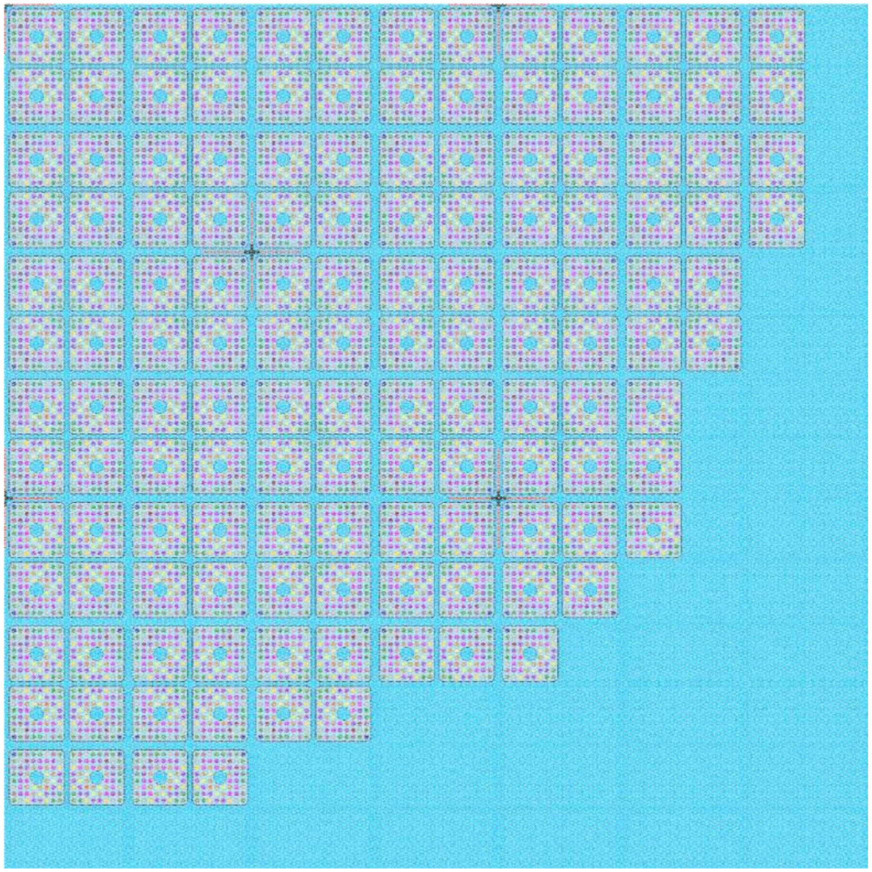
- Note only in CASMO-4E or CASMO-5M



CASMO Multi-assembly Capabilities



PWR

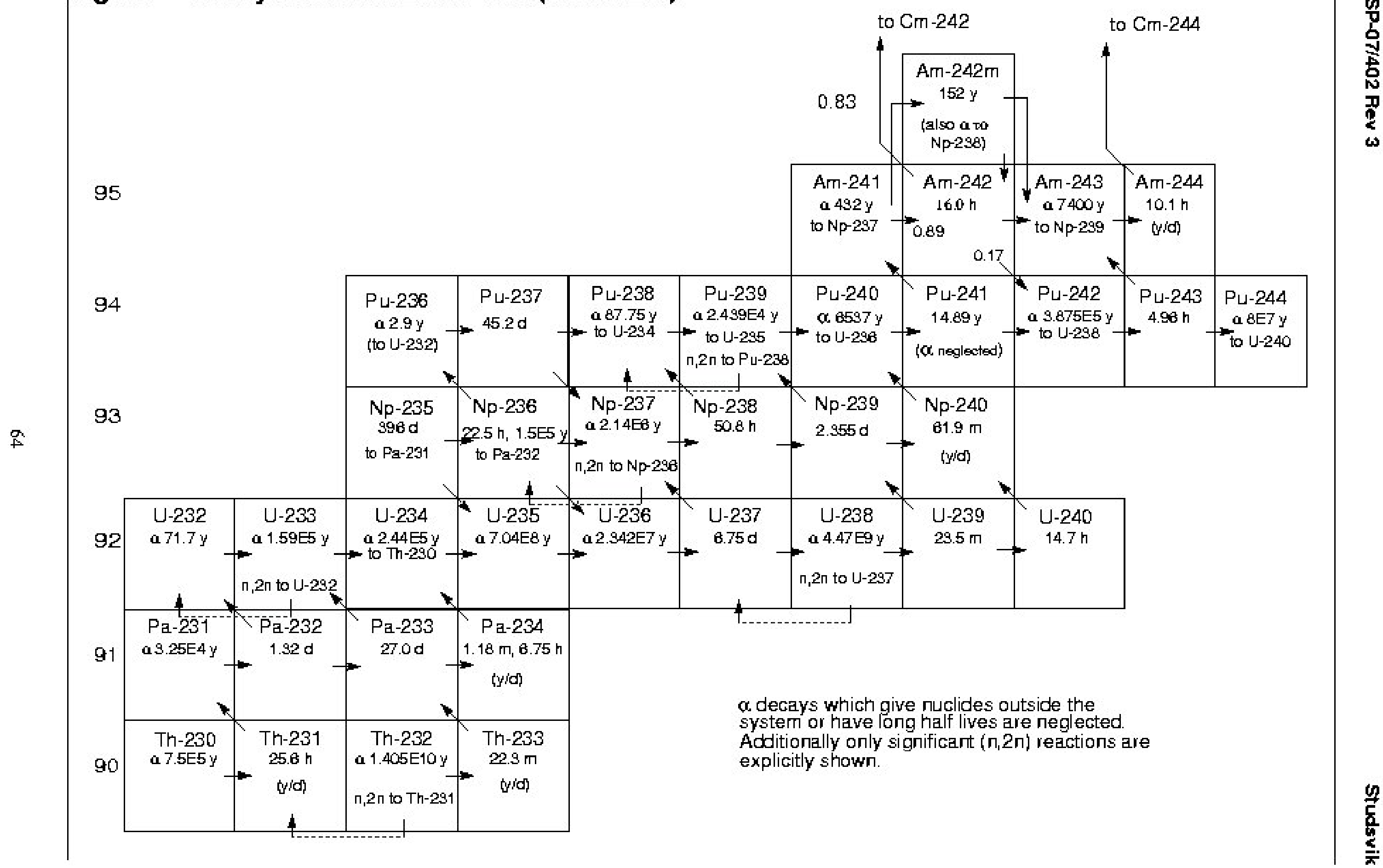


BWR

Large 2-D problems can be solved, if bundle/rack pitch is uniform.
(Used extensively for verification of downstream nodal codes)

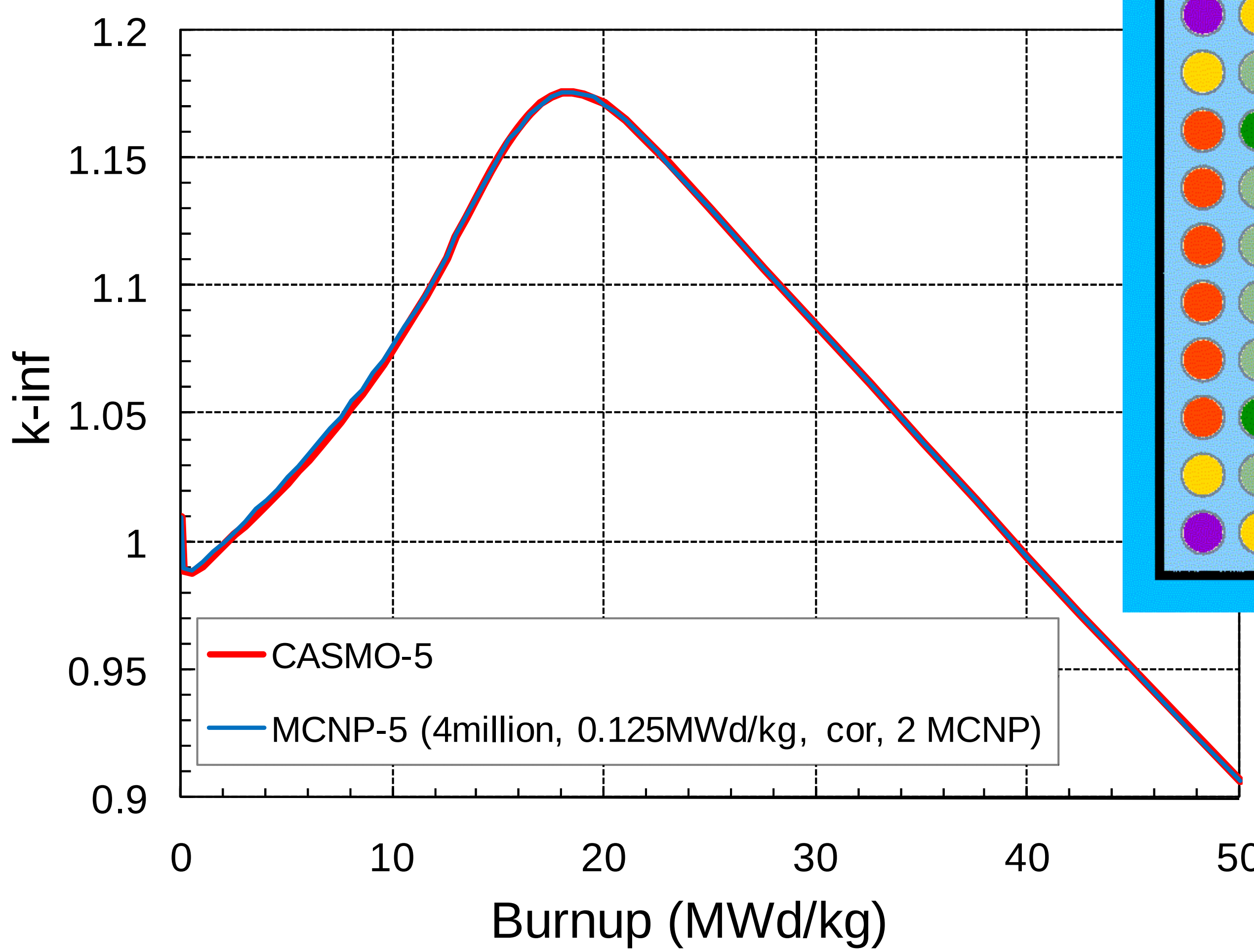
CASMO Actinide Depletion Chains

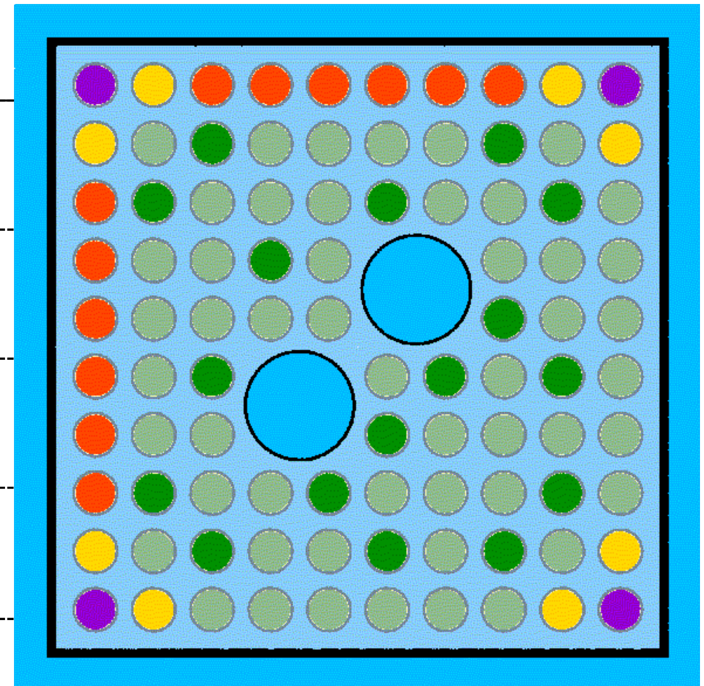
Fig. 1.1 Heavy Nuclides: Lower Part (to Am-244)



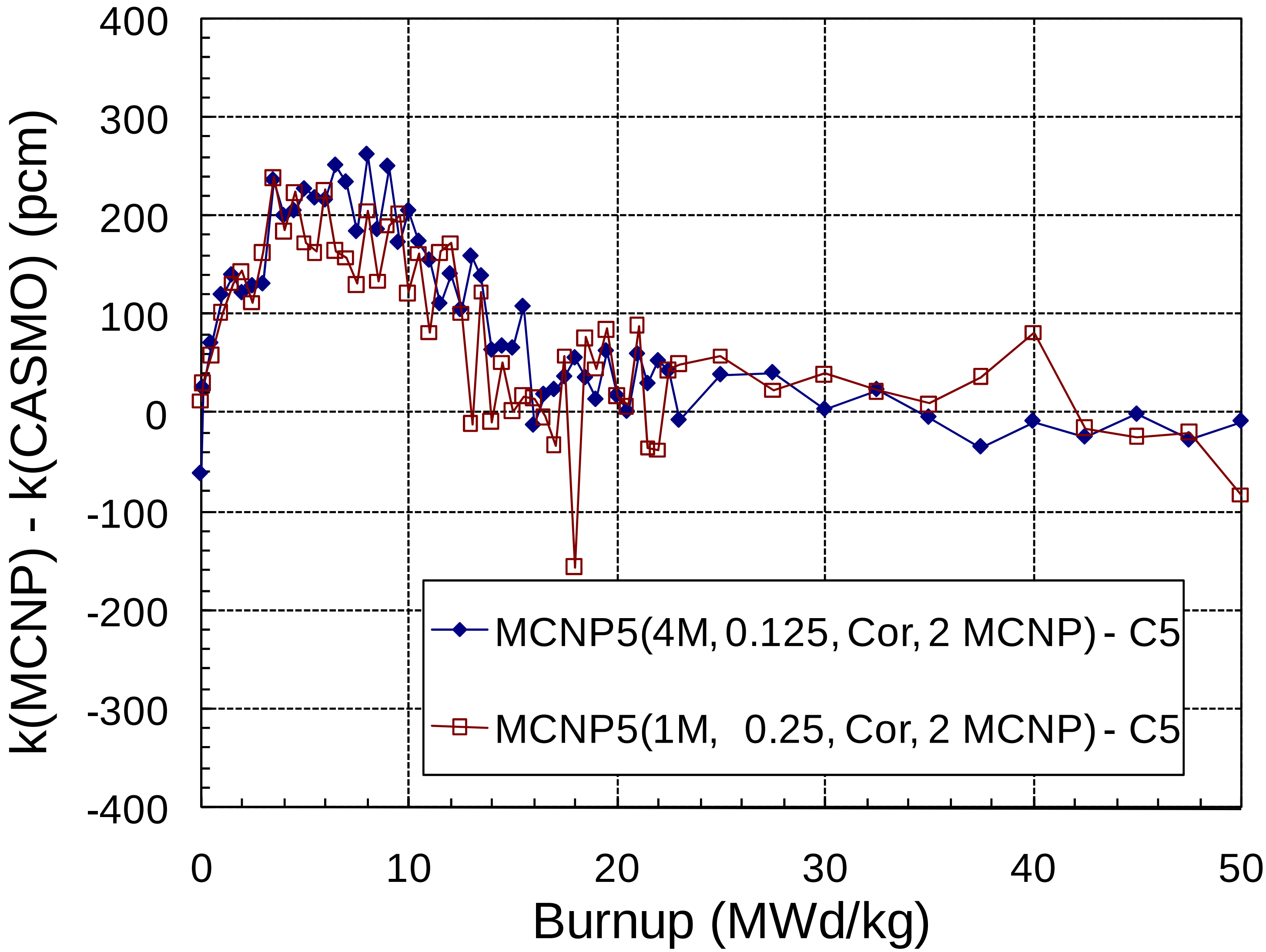
No Lumped Fission Products in E6/J2/E7 Libraries

BWR Bundle Depletion
CASMO vs. MCNP/ORIGEN

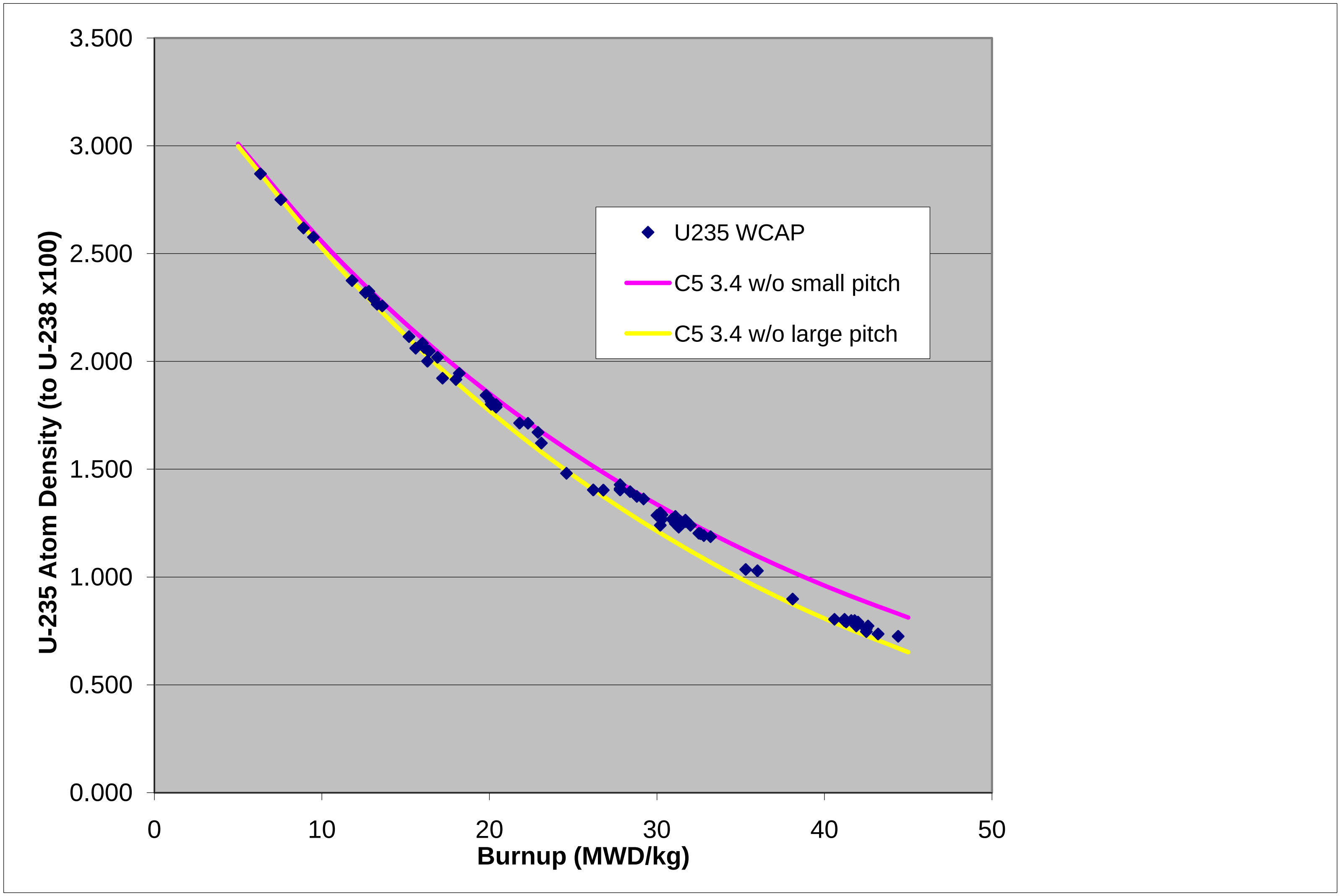




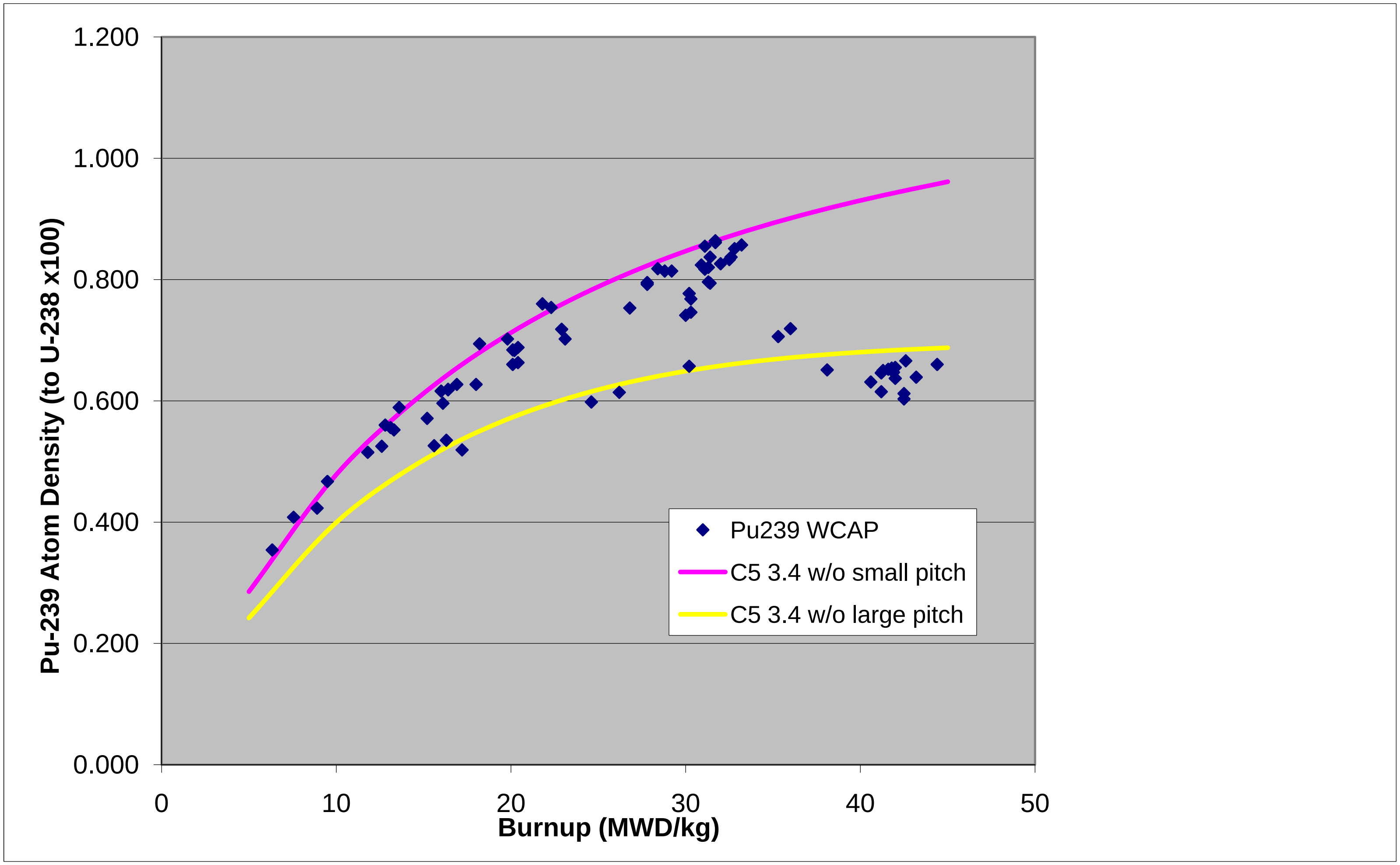
+/- 200 pcm Difference with Depletion



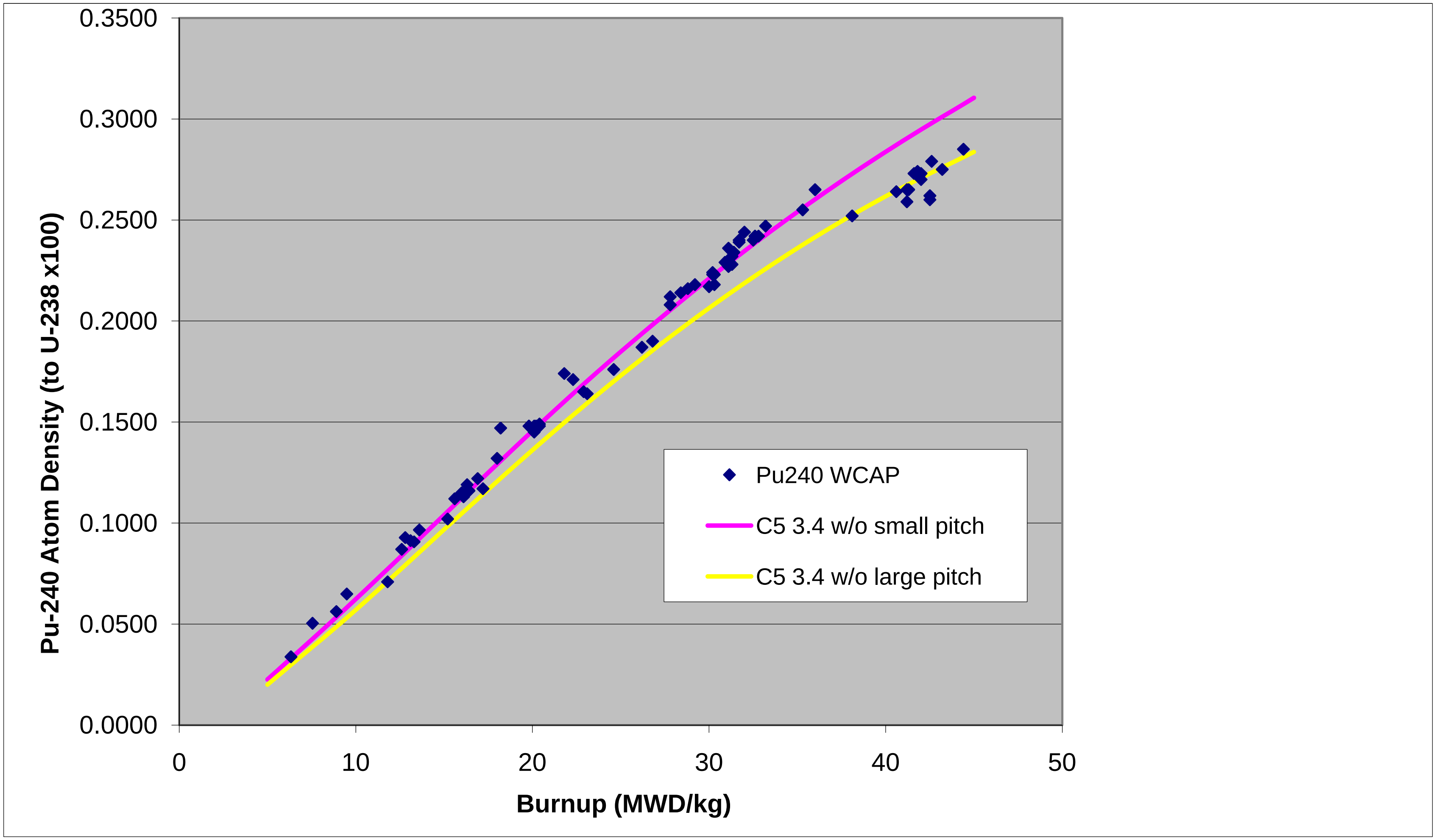
Measured PWR Isotopics – U-235



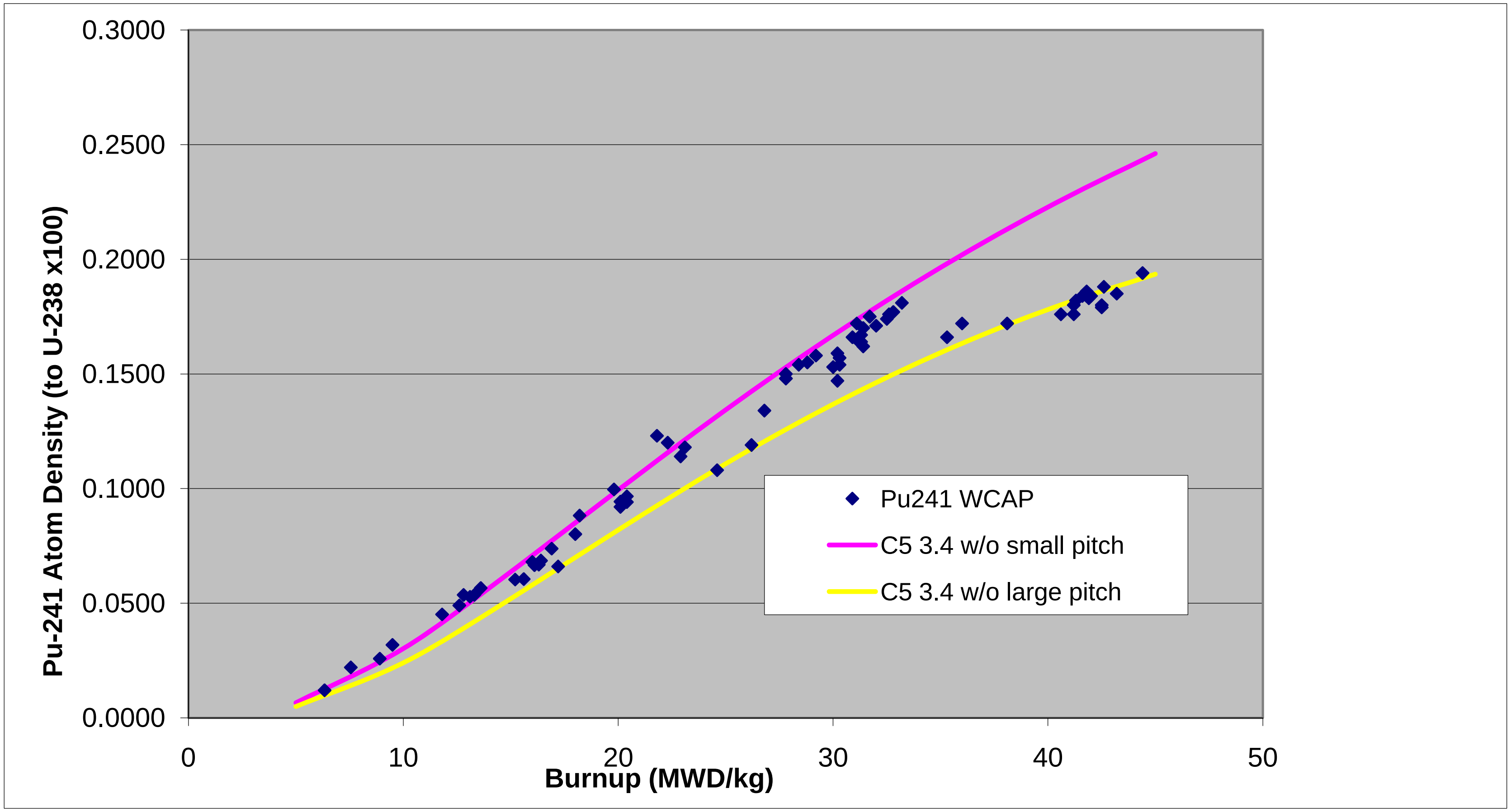
Measured PWR Isotopics – Pu-239



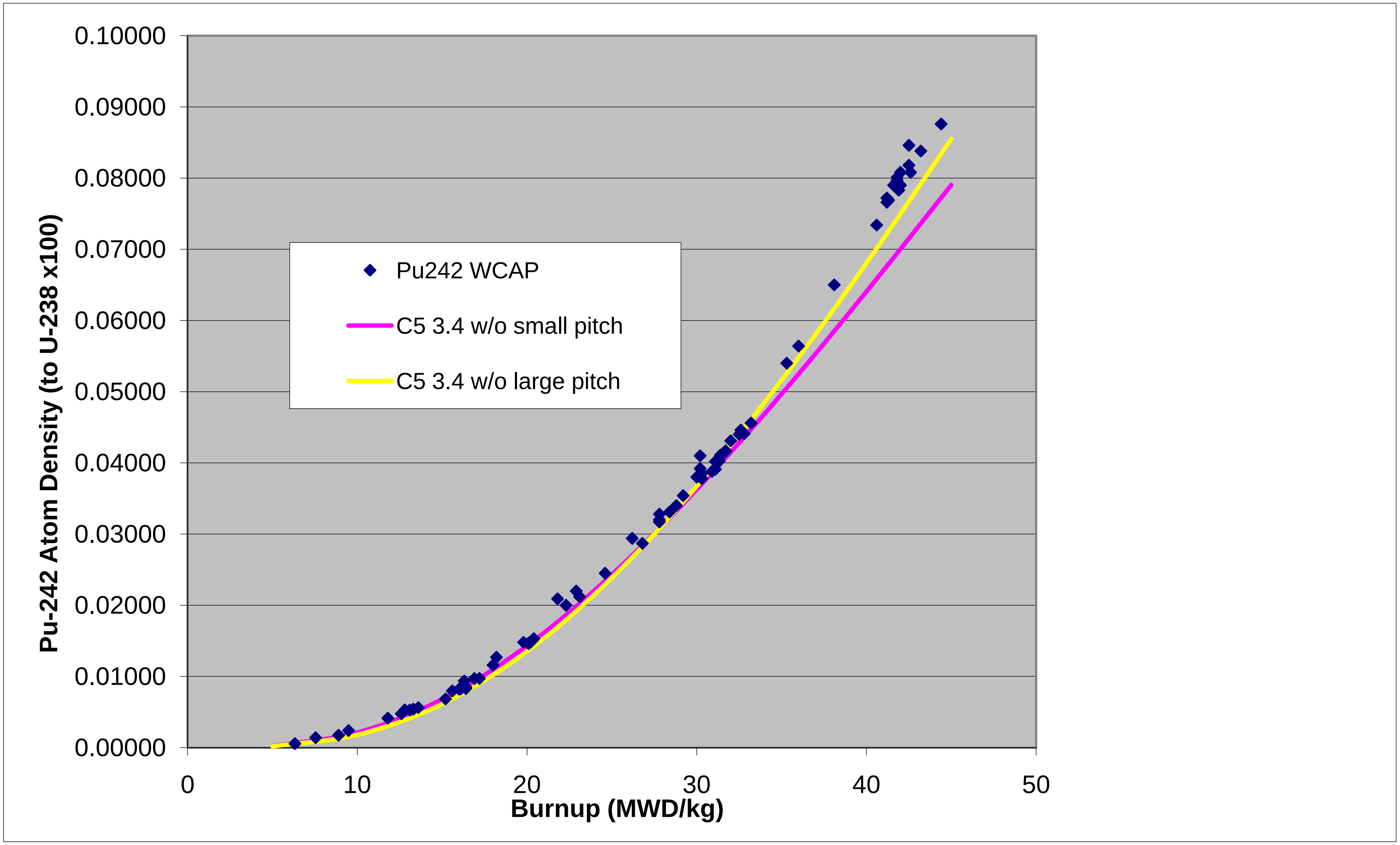
Measured PWR Isotopics – Pu-240



Measured PWR Isotopics – Pu-241



Measured PWR Isotopics – Pu-242

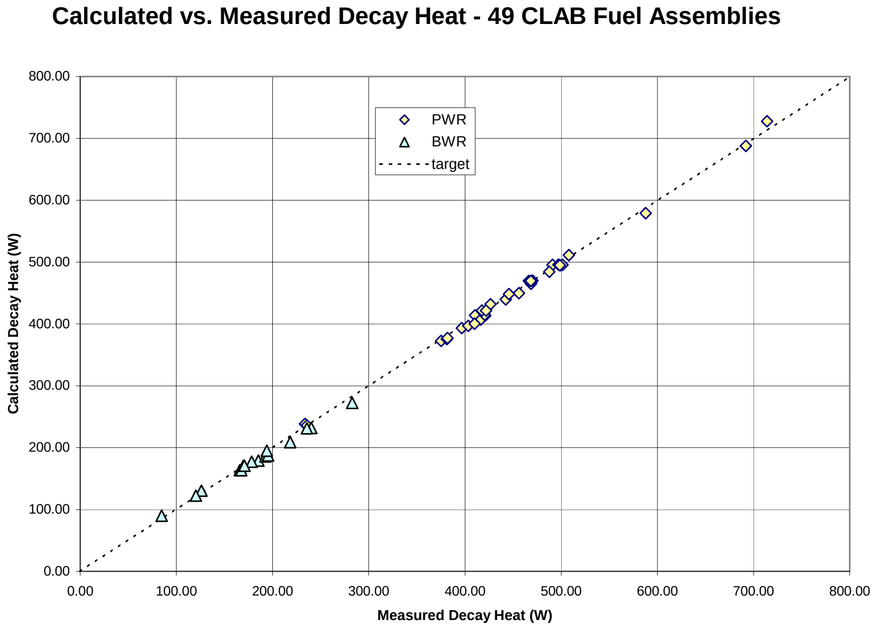


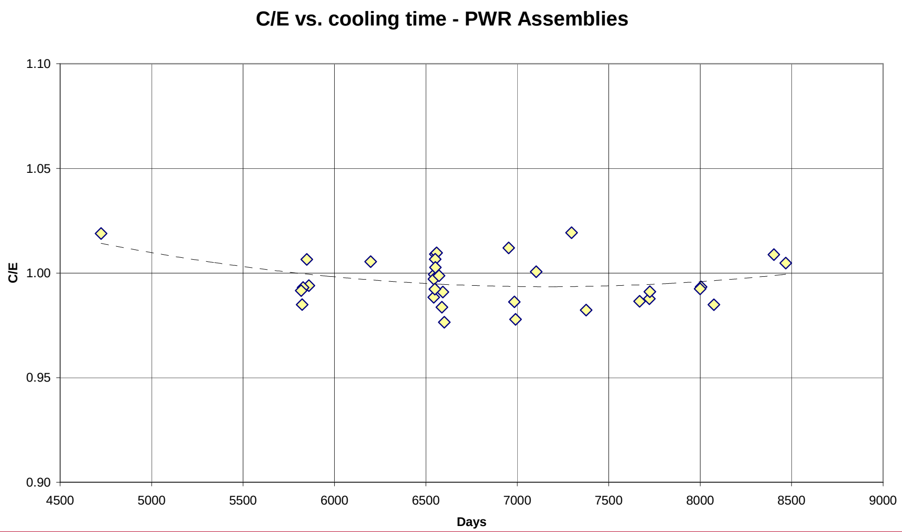
CASMO 17x17 PWR Depletion Comparisons

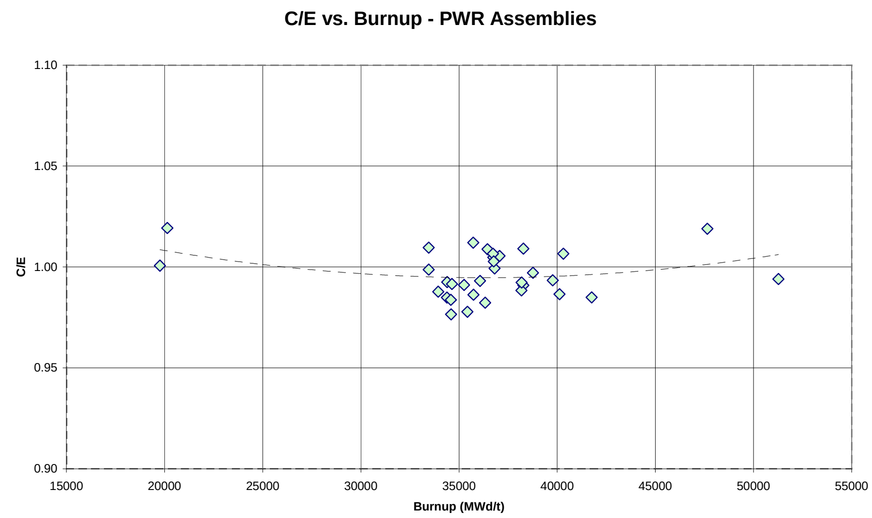
	(Mwd/kg)	C3/E4	C4/E4	C4/E6A	C4/E6U	C4/J2	C5/E7
Bor=500	0.1	1.33186	1.3313	1.3322	1.3266	1.3356	1.33346
	40	0.99961	1.0002	0.9944	0.9931	0.9977	0.99878
	80	0.78065	0.7863	0.7794	0.7837	0.7856	0.78936
Reactivity 0.1 to 40	pcm	33225	33105	33782	33348	33797	33468
Reactivity 0.1 to 80	pcm	55121	54493	55277	54292	55004	54410
% difference in decrement	(0.1 – 40)	-0.73	-1.08	0.94	-0.36	0.98	Reference
% difference in decrement	(0.1 – 80)	1.31	0.15	1.59	-0.22	1.09	Reference

- **Code versions, energy group structures, and nuclear data libraries change reactivity decrements only a small amount relative to Kopp's recommendation of 5% conservatism.**

Fission Product Benchmarking: Decay Heat







Typical PWR Core-Follow Results

	CASMO/SIMULATE minus Measured									
	HFP Boron (ppm)		HZP Boron (ppm)	Radial TIP		Axial TIP		Node-wise TIP	
	Mean	St Dev	Mean	Mean	St Dev	Mean	St Dev	Mean	St Dev
Cycle 1	-28	10	-22	1.4%	0.3%	3.0%	0.6%	3.9%	0.5%
Cycle 2	-19	6	-6	1.2%	0.3%	4.3%	1.9%	5.1%	2.0%
Cycle 3	7	8	18	1.2%	0.4%	3.0%	0.5%	3.9%	0.6%
Cycle 4	11	12	23	1.3%	0.4%	3.6%	1.4%	4.7%	1.4%
	-5	**19**	**3**	**1.3%**	**0.4%**	**3.5%**	**1.3%**	**4.4%**	**1.3%**

- **Power distributions are very accurately predicted.**
- **HFP core reactivity with depletion predicted +/- 300 pcm over all cycles.**
- **HZP to HFP reactivity (Doppler plus MTC) are well predicted.**

- **Net effects of many complex factors are implicitly included in comparisons. (e.g., clad oxidation, clad thinning, fuel cracking, rod bowing, etc.)**

Typical BWR Core-Follow Results

15 cycles of operation:

- **160 hot data points**
- **100 cold data points**

CASMO/SIMULATE minus Measured										
	Hot K-eff (pcm)		Cold k-eff (pcm)		Radial TIP		Axial TIP		Total TIP	
	Mean	RMS	Mean	RMS	Mean %	RMS	Mean %	RMS	Mean %	RMS
SIMULATE 2-group	70	169	240	292	1.97	0.56	2.44	0.61	3.75	0.56
SIMULATE 4-group	-26	172	200	300	1.98	0.54	2.50	0.54	3.78	0.53

- **Power distributions are very accurately predicted.**
- **HFP core reactivity with depletion predicted +/- 300 pcm over cycles.**
- **Cold to HFP reactivity (Doppler plus MTC) predicted +/- 300 pcm.**

- **Net effects of many complex factors are implicitly included in comparisons. (e.g., clad oxidation, clad thinning, fuel cracking, rod/channel bow, etc.)**

Summary

- **Criticals comparisons demonstrate that CASMO accurately predicts BOL bundle reactivity.**

- **Storage rack/flux trap criticals comparisons demonstrate that CASMO can accurately predict simulated cold SFP configurations.**

- **MCNP/ORIGEN depletion comparisons demonstrate that CASMO can accurately predict depletion reactivity effects (for known nuclear data libraries).**

- **Spent fuel isotopics comparisons demonstrate that CASMO can accurately predict actinide buildups and burnout rates.**

- **PWR and BWR core follow results demonstrate that CASMO accurately predicts HFP core depletion effects.**

- **BWR cold criticals comparisons demonstrate that CASMO accurately predicts cold depleted fuel reactivity.**

- ***In-core criticality is predicted with little dependence on core burnup and with uncertainties much smaller than Kopp's 5% conservatism.***

NRC Topical Reports for CASMO/SIMULATE

- **Yankee Atomic Electric Company, "CASMO-3G Validation," *YAEC-1363, April, 1988.***
- **Yankee Atomic Electric Company, "SIMULATE-3 Validation and Verification, *YAEC-1659,* September, 1988.**
- **TU Electric Co., "Steady State Reactor Physics Methodology," *RXE-89-003-NP, July, 1989.***
- **Southern California Edison Co., "PWR Reactor Physics Methodology Using CASMO-3/SIMULATE-3," *SCE-9001-A, September, 1992.***
- **Duke Power Company, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," *DPC-NE-1004A, November, 1992***
- **Entergy Operations, Inc., "Qualification of Reactor Physics Methods for the Pressurized Water Reactors of the Entergy System," *ENEAD-01-NA-A REV 0, December, 1993.***
- **Omaha Public Power District, "Neutronics Design Methods and Verification," *OPPD-NA-8302-NP REV 4, May, 1994.***
- **TU Electric Entergy Operations, Inc., "Verification of CECOR Coefficient Methodology for Application to Pressurized Water Reactors of the Entergy System," *ENEAD-02-NP-A REV 0, September,* 1994.**
- **Arizona Public Service Company, "PWR Reactor Physics Methodology Using CASMO-4/SIMULATE-3, September, 1999**
- **Northern States Power, Prairie Island Nuclear Power Plant, "Qualification of Reactor Physics Methods for Application to Prairie Island", *NSPNAD-8101-A Revision 2, October 2000.***
- **Dominion, North Anna and Surry, "Qualification of the Studsvik Core Management System Reactor Physics Methods for Application to North Anna and Surry Power Stations, DOMNFA-1-Rev. 0.0 -A, June 2003**

Studsvik

Using ANSI/ANS-8.27 for Burnup Credit Validation

Dale Lancaster

Chairman of 8.27 Working Group

NuclearConsultants.com

and

Charles T. Rombough

Secretary of 8.27 Working Group

CTR Technical Services, Inc

Introduction

- Burnup Credit Standard started early 2002
- Approved as ANSI/ANS-8.27-2008 on August 14, 2008
- Large working group met twice a year
- Covers Pools, Casks, and Disposal
- Limited to Commercial PWRs and BWRs

Working Group Members

D. B. Lancaster (Chair), *NuclearConsultants.com*
C. T. Rombough (Secretary), *CTR Technical Services, Inc.*

S. Anton, *Holtec International*
R. Beall, *Constellation Energy*
R. A. Hommerson, *Individual*
J. R. Massari, *Constellation Energy*
P. Narayanan, *TransNuclear Inc.*
M. Rahimi, *U. S. NRC*
G. R. Walden, *Duke Power*
A. Zimmer, *General Atomics*
A. C. Attard, *U. S. NRC*
M. C. Brady Raap, *PNL*
L. I. Kopp, *Individual*
R. D. McKnight, *ANL*
C. V. Parks, *ORNL*
D. A. Thomas, *AREVA*
A. H. Wells, *EPRI*
J. F. Zino, *GE Nuclear Energy*
S. P. Baker, *TransWare*
J. P. Coletta, *Duke Power*
Z. Martin, *TVA*
D. Mennerdahl, *Sweden*
H. Pfeifer, *NAC International*
S. E. Turner, *Individual*
C. J. Withee, *U. S. NRC*

The following is a list of people who supported the working group but were not able to actively participate throughout the entire process:

J. Boshoven, D. Cacciapouti, M. DeHart, M. DeVoe, D. Galvin, J. Gulliford, L. Hassler, D. Hutson, R. Jones, R. Kunita, A. J. Machiels, L. Markova, M. Mason, S. Mitake, D. Mueller, G. O'Connor, P. O'Donnell, H. Toffer, J. C. Wagner, C. Walker, B. Wilson.

8.27 Table of Contents

(Subsections not used today are not listed)

Criteria to establish subcriticality

$$k_p + \Delta k_p + \Delta k_i + \Delta k_b \leq k_c - \Delta k_c - \Delta k_x - \Delta k_m$$

k_p is the calculated multiplication factor

Δk_p is an allowance for uncertainties in the determination of k_p

Δk_i is an allowance for the bias and uncertainty in k_p due to depletion uncertainty in the calculated nuclide compositions.

Δk_b is an allowance for uncertainty in k_p due to uncertainty in the assigned burnup value.

k_c is the multiplication factor that results from the calculation of the benchmark criticality experiments.

Δk_c is an allowance for uncertainty in k_c

Δk_x is a potential supplement to k_c and/or Δk_c that may be included to provide an allowance for the bias and uncertainty from nuclide cross section data that might not be adequately accounted for in the benchmark criticality experiments used for k_c.

Δk_m is a margin for unknown uncertainties

Last Paragraph of Section 4 of 8.27

"In one method of validation, Δk_i and Δk_x are inseparable and are determined together. (See Sec. 5.2.)"

We will use this for PWR pool analysis.

5 Validation for burnup credit

"The validation of the burnup credit methodology may be accomplished by validation of each analysis component (i.e., analysis to determine the nuclide composition and analysis to determine the neutron multiplication factor) ***<u>or by a combined validation approach</u>*** *where the bias and uncertainty terms from the individual analysis components are not determined individually."*

Section 5.1 discusses validation by components. PWR pools will use Section 5.2.

5.2 Combined validation approach

"Validation of the burnup credit models (i.e., determination of nuclide composition and neutron multiplication factor) may be performed by analysis of applicable critical systems consisting of irradiated fuel with a known irradiation history. For this method of validation, the terms Δk_i , Δk_x, and potentially parts of k_c can be inseparable. The uncertainty in the isotopic content and cross sections is captured in the calculation of the multiplication factor of the criticality experiment with irradiated fuel."

Using the Fuel Management Experience For Validation

- In the combined validation approach, experimental data with spent nuclear fuel is needed.
- The experimental data is the fuel management experience (regular measurements of critical ppm, power distributions, and reactivity coefficients).

Current PWR Reactor Analysis

- 2D Lattice Codes
  - PHOENIX (PARAGON)
  - CASMO, etc.
- 3D Nodal Code
  - ANC
  - SIMULATE
  - ROCS, etc.

Current PWR Reactor Analysis

- Criticality predictions at startup within 50 ppm acceptance criteria (approx. 0.004 in k)
- End of Cycle hot full power predictions on average are equivalent to hot zero power startup

Current PWR Reactor Analysis (continued)

- The PWR Tech Specs require reactivity agreement with prediction of better than 1% in k.
- Core depletion does not significantly affect the predicted agreement

Current BWR Reactor Analysis

- The BWR Tech Specs require reactivity agreement with prediction of better than 1% in k.
- BWR cold criticals, which include fission products, are in good agreement
- BWR cold criticals resemble a spent fuel pool geometry

Conclusion of Validation

- Depletion does not significantly affect core reactivity calculations (typically significantly less than 1%)
- 5% of reactivity decrement inherently assumes that depletion affects reactivity calculations

Conclusion of Validation (continued)

- Therefore, 5% of reactivity decrement is very conservative
- The 5% of reactivity decrement inherently covers the lack of fission product criticals and other core operating effects
- A depletion uncertainty less than 5% could be justified in an application

What does the 5% cover

- The 5% covers the change in isotopic content and the worth of the new isotopes (fission products and actinides).
- No benchmarking of the fission products is needed since it is covered by the 5% uncertainty in the depletion delta k.
- Critical Experiments are still needed to validate the initial condition (UO_2)

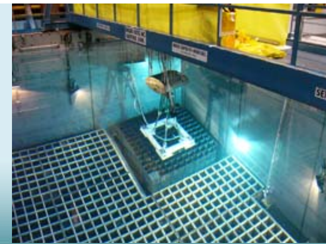
Rest of Standard

- Sections 5.3, 6, and 7 do not raise any issues worthy of discussing at this point

Summary

- ANSI/ANS-8.27 is released and covers spent fuel pools
- 5% of the delta k of depletion can be used as a conservative uncertainty of the depletion analysis.
- The uncertainty is justified by power reactor measurements.
- This conservative uncertainty covers all validation issues beyond UO2 fresh fuel conditions.






Spent Fuel Wet Storage Criticality Analysis
Margin versus Conservatism

Kristopher Cummings
May 1, 2009

Agenda

- Definitions
- Imposed Margins
- PWR Rack Conservatisms
- BWR Rack Conservatisms
- Minor Reactivity Effects
- Complex Applications
- Conclusions

Definitions

HOLTEC INTERNATIONAL

- Margin: The amount by which the result is below the specified limit.
- Conservatism: Assumptions or techniques used in the methodology or analysis which ensure that the calculated reactivity is less than the actual reactivity.

Imposed Margins

- What is the imposed margin in the analysis?
  - Normal conditions
    - No soluble boron:
      - No administrative margin (1.0)
      - Soluble boron margin 0.16 Δk (2000ppm Tech Spec)
    - Soluble boron credit
      - Administrative margin 0.05 Δk (0.95)
      - Soluble boron margin 0.10 Δk (600ppm credited)
  - Accident conditions, fresh fuel assembly misload
    - Administrative margin 0.05 Δk (0.95)
    - Soluble boron margin 0.06 Δk (1200ppm credited)
  - Additional Δk margin may be applied as target k_{eff}

PWR Rack Conservatisms

- Possible conservatisms in PWR rack calculations could be:
  - Reference bounding fuel assembly (0.01 Δk)
  - No credit of IFBA, Erbium, Gd_2O_3 for fresh fuel (0.1 Δk)
  - Bounding depletion parameters (fuel temp, moderator temp, soluble boron, power density)
    - Moderator temperature (0.005 Δk)
    - Soluble boron (0.005 Δk)
  - Modeling of fuel inserts during depletion - bounding insert in all fuel assemblies over entire active length (0.01 Δk compared to no inserts)
  - Axial burnup distribution
    - Bounding distributed profile (0.010-0.015 Δk for high-burnup)
    - Flat/Uniform profile (0.005 Δk for low burnup)

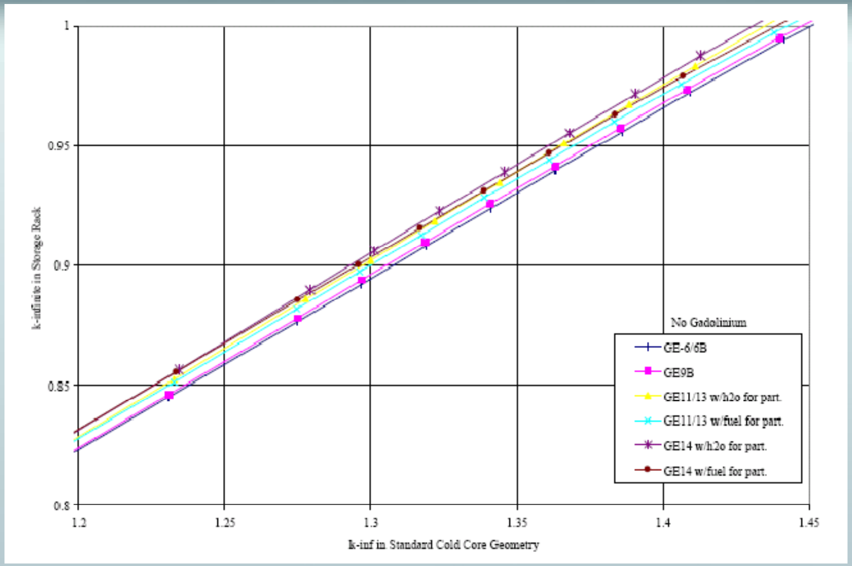
PWR Rack Conservatisms

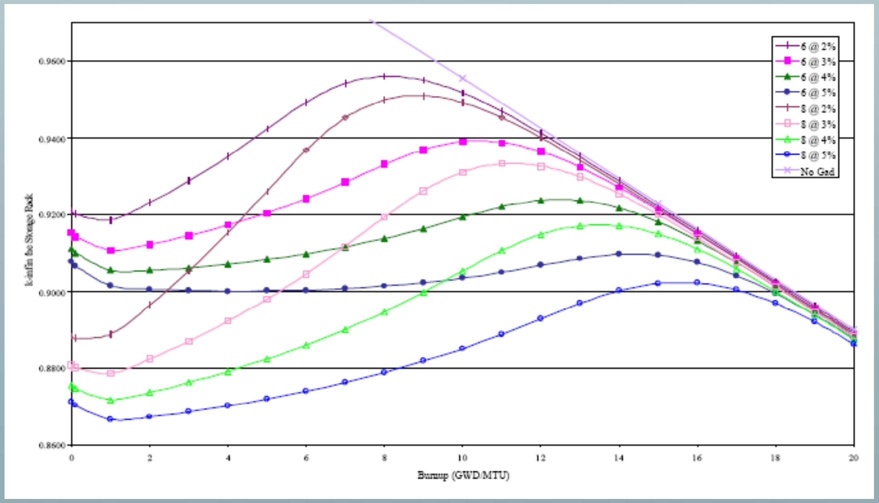
- Possible conservatisms in PWR rack calculations could be:
  - Zero cooling time (0.04 Δk for 20 years cooling)
  - Fuel inserts not credited in SFP (0.005 Δk for spent WABAs, BPRAs)

BWR Criticality Analyses

HOLTEC INTERNATIONAL

- Several independent acceptance criteria:
  - 1) Maximum enrichment, no gad, no burnup
  - 2) K_{inf} in the standard cold core geometry
  - 3) Minimum gad rods at minimum loading





BWR Rack Conservatisms

- Possible conservatisms for various BWR rack acceptance criteria could be:
  - Maximum fresh assembly enrichment (~3.3 wt% ^{235}U)
    - No credit for Gadolinium (0.02 Δk)
    - No credit for burnup
  - K_{inf} in the standard cold core geometry
    - Reference bounding fuel assembly (0.01 Δk)
    - Maximum reactivity fuel assembly (0.17 Δk)
    - Bounding depletion parameters (fuel temp, moderator temp, power density)
      - Moderator temperature (0.005 Δk)
  - Lower bound number of Gadolinium rods at lower bound loading.
    - Maximum reactivity fuel assembly
    - No credit for reactivity decrement past peak burnup (0.15 Δk)
    - No credit for additional Gadolinium rods (0.02 Δk)

Additional Conservatisms

HOLTEC INTERNATIONAL

- Some conservatisms may exist but can't be credited:
  - Radial leakage, except for peripheral cells.
  - Additional burnup of actual fuel assemblies above required amount.
  - Residual fixed neutron absorber if not credited and surveyed (i.e., loss of Boraflex)

Minor Reactivity Effects

- Some issues have small reactivity effects and can be covered by conservatisms and margin without explicit calculations:
  - Neglecting grid straps (conservative upto 1500ppm)
  - Eccentric positioning in the storage cell. (negligible for racks with neutron absorber)
  - Slight modeling differences for simplicity (<0.001 Δk)
  - Effect of soluble boron on manufacturing tolerances (negligible even for 2000ppm credit)
  - Tolerances in burnable poisons, fuel inserts, etc. (covered by bounding depletion parameters)
  - Some fuel tolerances (guide tube thickness, clad inner dimension, instrument tube dimensions, < 0.001 Δk)

Complex patterns

HOLTEC INTERNATIONAL

- What effect do complex patterns have on the conservatism and margin:
  - The margin is typically not affected:
    - Same administrative margin
    - Approximately the same soluble boron credited
  - The conservatism may be affected:
    - Consideration of blanketed versus non-blanketed assemblies.
    - Cooling time credited
    - Assembly designs considered seperately
  - More complex patterns (checkerboards, multiple loading patterns, etc) do not necessarily reduce either the margin or the conservatism.

Conclusions

- *Conservatisms* have been modified as more sophisticated patterns and methodologies have been implemented.
- As more complex patterns are employed some excessive conservatisms are reduced to provide usable loading patterns in the spent fuel pool.
- The recent items of interest to NRC (grid spacers, eccentric positioning, effect of soluble boron) have a small reactivity effect compared to the *margins* to safety.
- There has been and remains to be significant *margin* to safety (subcriticality) in all spent fuel pool criticality calculations.

Fuel Assembly Misloading

Prepared by Ed Knuckles FP&L

Planning for Storage Configurations

Based on:

- Predicted EOC assembly burnup (BU)
- Core calculations/incore measurements
- Associated with the actual core operation
- Controlled by Q/A Program
- Short window BU used for conservatism
- Redone if shutdown conditions change

Storage Configurations (example)

	1	2	3	4	5	6	7	8
R	NN27	NN30	NN31	NN32	NN34	NN38	NN41	NN42
S	W57	W60	V05	V08	V09	V10	NN01	NN02
T	W15	W16	W17	W18	W23	W24	W25	W32
U	Z54	Z55	Z58	Z60	Z63	Z64	X01	X11
V	Z13	Z15	Z25	Z26	Z27	Z29	Z30	Z31

	60	61	62	63	64	65	66	67
AA	X13	Z04	NN33	PP31	L41	P49	P51	PP26
BB			Z45	PP25	L16U4	Z16	X18	PP30
CC			Z48	X20	Z20	PP40	P13	U15
DD			Z02	X02	Z28	PP51	N39	S15
EE			NN36	RR18	SS11	PP54	N13	T15

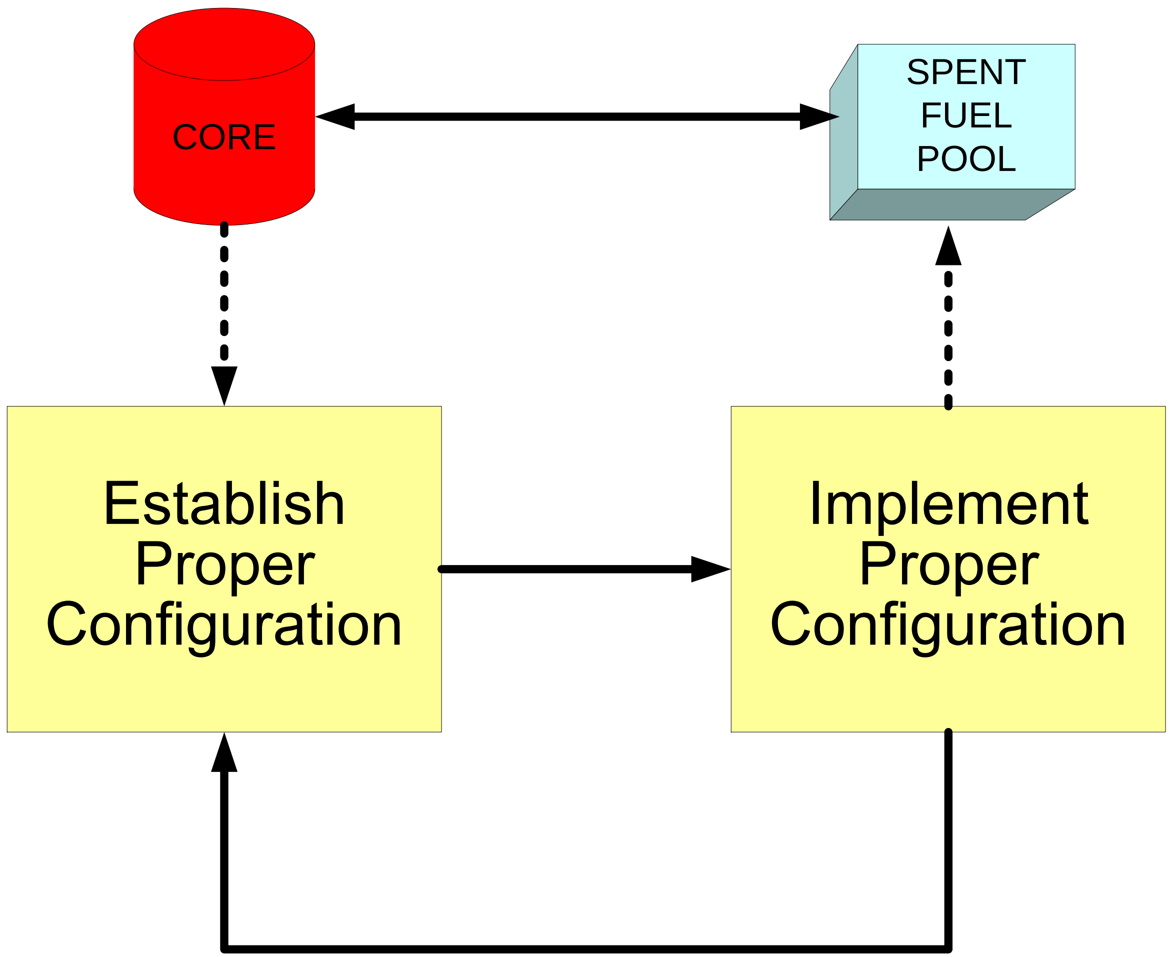
Complexity

- Blanketed vs. Unblanketed
- Poison Inserts
  - 2 x 4
  - 1 x 4
  - 0 x 4
- Interface
  - Between configurations
  - Between regions
  - Between rack & wall

Establishing Proper Configuration

- Standards/procedures can provide requirements
- Directs the source of the information to be used
- Technical Specifications (TS) contain the BU credit information
- Assembly assigned to a storage configuration based on its initial enrichment and BU
- Document engineering results & verify
- Transmit results to Plant

Configuration Control



Fuel Move Procedure

- Sequence of physical operations
- Based on engineering results
- Plant prepares the fuel move procedure
- Assembly assigned a location in the spent fuel pool (SFP)
- Designated by an alpha-numeric cell ID
- ID same as signage on wall in the SFP
- Review and approve fuel move procedure

Fuel Assembly & Insert Shuffle Control

STEP NUMBER	LATCH DATE	LATCH TIME	REMOVE FROM	INSERT IN	UNLATCH TIME	INITIALS	ASSEMBLY NUMBER	INSERT NUMBER
108A	3/14/06	0042	REACTOR CORE N12	CAVITY UPENDER	0048	[illegible]	AC04	
108B	3/14/06	0054	SFP UPENDER	SFP L24	0058	[illegible]	AC04	
109A	3/14/06	0052	REACTOR CORE D13	CAVITY UPENDER	0054	[illegible]	AC05	
109B	3/14/06	0104	SFP UPENDER	SFP L26	0108	[illegible]	AC05	
110A	3/14/06	0103	REACTOR CORE E13	CAVITY UPENDER	0109	[illegible]	AE21	
110B	3/14/06	0114	SFP UPENDER	SFP MT93	118	[illegible]	AE21	

Operation & Control

- Movement of every fuel assembly controlled by plant procedure.
- Requires:
  - Move director in control room (reload),
  - Fuel crane operator,
  - SFP supervisory oversight, and
  - Upender operator (reload only)
- Transfer of assembly within the pool/to containment

Operation & Control (continued)

- Independent direction of each step in move sequence
  - From approved procedure
  - In parallel with the operator who has a copy of procedure
- Independent verification & documentation
  - Time move sequence initiated,
  - Time move sequence completed, and
  - Initials completion.

Communication

- Continuously between supervisor and crane operator
- Three way communication
  - Directs operator to location in pool
  - Assures order in move sequence:
    - Understood by the operator and
    - supervisor acknowledges operator correctly understood order
- Supervisor directs operator to lower/latch assembly in the cell.
- Operator informs supervisor at bottom of cell & ready to unlatch/latch assembly
- Supervisor directs operator to unlatch/latch assembly

Barriers to Misloading

- Technical Specifications
- Standardization
- Q/A Program
- Procedurized Evolution
- Slow Evolution
- Physical Indexing
- Three Way Communication
- Spotters Verify Indexing
- Physical Inventory to Verify Location

Factors Affecting Barriers

- **Technical Specifications**: Clarity, Compliance
- **Standardization**: Clarity, Compliance
- **Q/A Program**: Compliance, Robustness
- **Proceduralized Evolution**: Compliance
- **Slow Evolution**: Attention to detail
- **Physical Indexing**: Cell spacing, Cell Pitch
- **Three Way Communication**: Clarity
- **Spotters Verify Indexing**: Refraction, Convection
- **Physical Inventory**: Piece count versus ID

Industry Operating Experience

- The nuclear industry through WANO and INPO routinely highlights fuel handling events
- Issuance of Operating Experience to plants
  - communicate lessons learned
  - causes, significance, and recommendations
  - incorporates lessons learned into the work practices
- INPO issued TR6-53 in 2006
  - Evaluation of fuel handling events between 2002 and 2005.
  - 10% of the 125 fuel handling related events were related to mispositioning of fuel or a fuel related component in the SFP
  - "Improper self-checking and verification practices contributed to 89% of the mispositioned components"

Summary

- Many barriers in place to mitigate the possibility of misloading
- INPO & WANO OE Reports help improve fuel handling work practices
- Verification, Self-Checking & Communication in all aspects is important in the movement process
- Physical process is slow enough to recognize errors
- Physical inventories prevent accumulation of misloadings

Conclusion

- Only a single item can be moved at a time
  - Fuel assembly with or without insert
  - Any other component moved in the pool
- Movement is a controlled process
- Timely recognition of misloadings
- Increasing complexity doesn't necessarily imply increasing
  - Probability of a misloading accident
  - Possibility of multiple misloading

NETCO
NORTHEAST TECHNOLOGY CORP

NRC Information Meeting

Reactivity Effects of Degraded Boraflex

5/1/09

Rockville, MD

NETCO

Overview

- Boraflex, borated silicone rubber product that provides reactivity hold-down in spent fuel pool
- Boraflex subject to time related environmental degradation
- There are several modes of degradation, modes broadly describable as uniform, or inhomogeneous or random (usually occur in combination)
- Degradation is measurable and predictable
- Without the benefit of realistic evaluation, calculated reactivity effects can significantly reduce criticality margins
- Realistic evaluation of associated reactivity effects (especially for inhomogeneous and random degradation) requires detailed calculational modeling in conjunction with prediction guided in-situ measurement

NETCO

Boraflex Degradation Modes

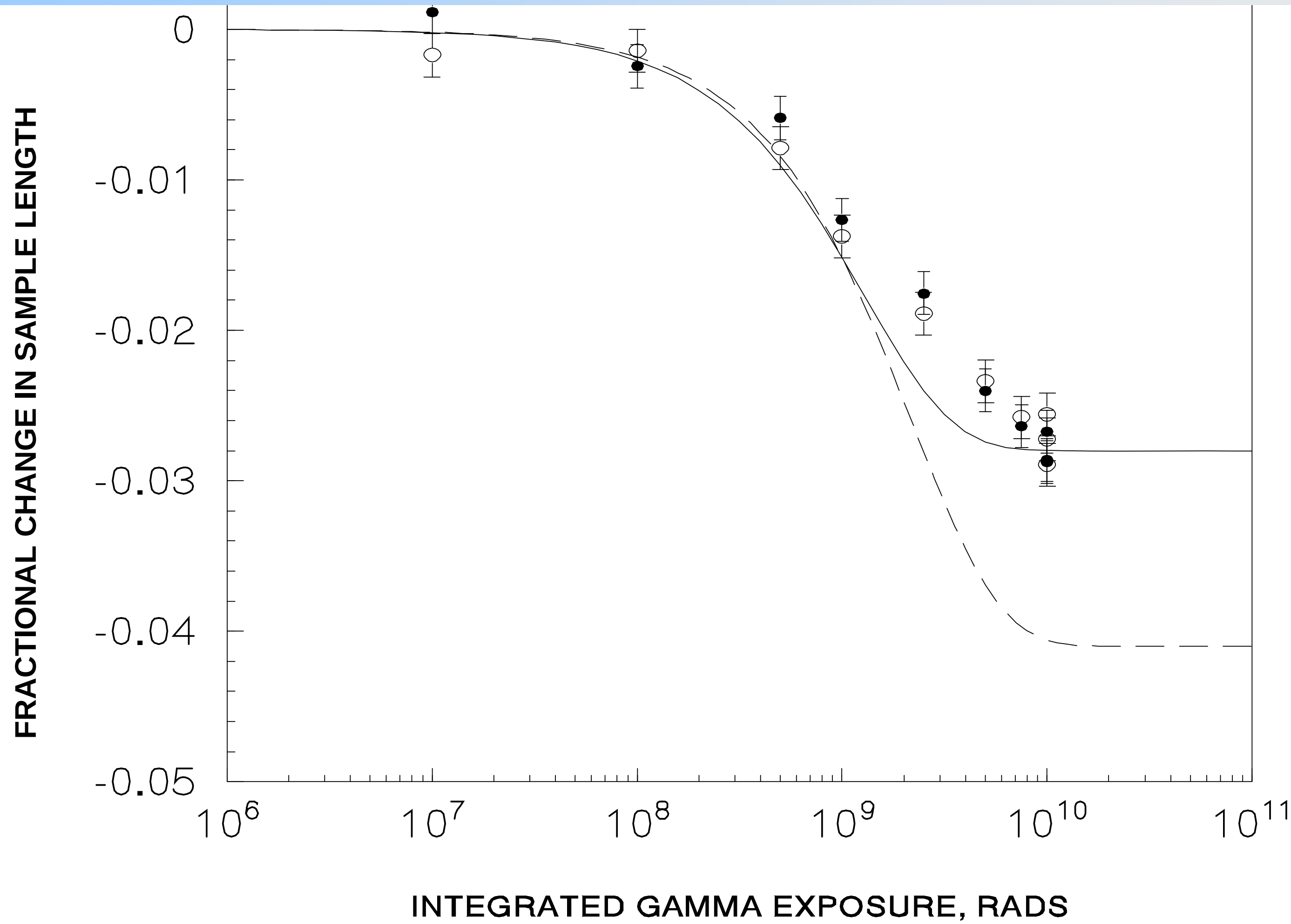
- Initially Boraflex matrix undergoes cross linking:
  - Increase in density
  - Potential for gap formation
  - Potential for end “pull back”
- At higher doses and in the presence of free oxygen Boraflex matrix undergoes chemical transformation to amorphous silica:
  - Potential for local dissolution and thinning
  - Potential for global dissolution and thinning

Boraflex Degradation Modes (continued)

- Densification and Shrinkage:
  - Early in Life Phenomena: Gaps form and can continue to grow in size
  - Saturates at an integrated gamma exposure of ~$1X10^{10}$ rads
- Dissolution:
  - Later in Life Phenomenon
  - Becomes a factor after cross-linking has saturated
  - Generally characterized as occurring very slowly
  - Local and General Dissolution affected by fuel rack design features
- Gaps and Densification: No loss of B-10 atoms
  - Boron redistributed
  - Results in non-uniform distribution of B-10 atoms
- Dissolution: Loss of B-10 atoms
  - Local effects tend to mitigate reactivity effects
  - Reactivity effects of generalized thinning - small

NETCO

Cross-Linking Induced Shrinkage of Boraflex

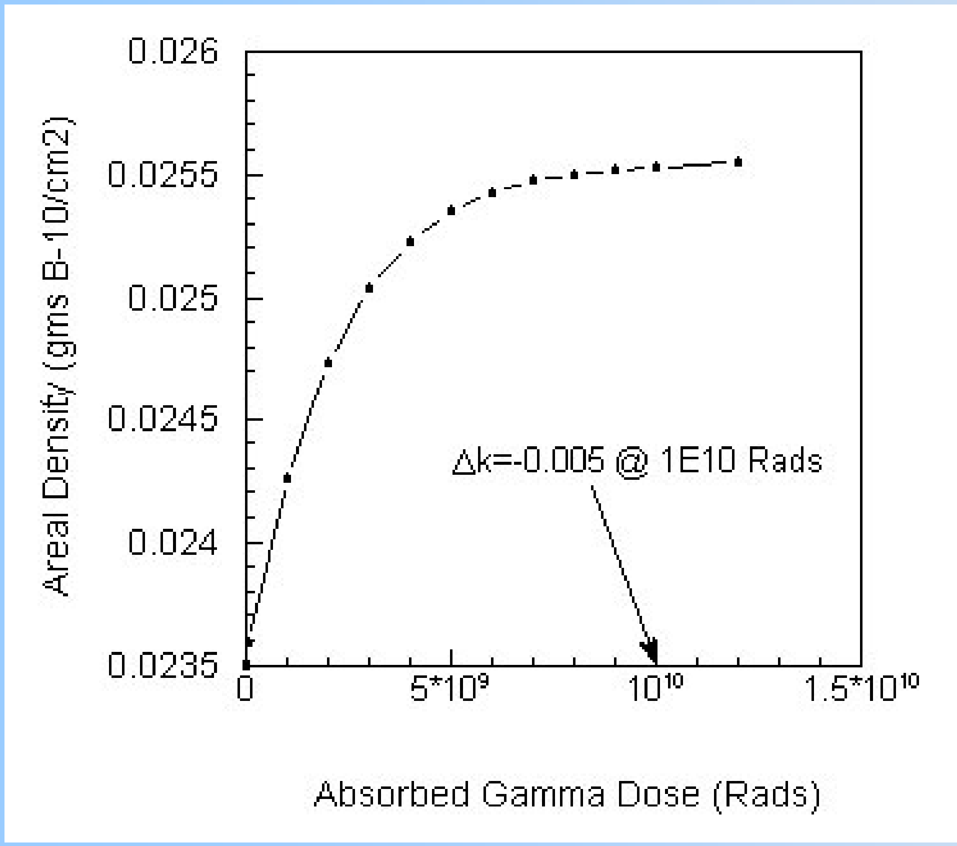


— UPPER BOUND SHRINKAGE PROJECTION
- LOWER BOUND SHRINKAGE PROJECTION

NETCO

Reactivity Effects of Degraded Boraflex

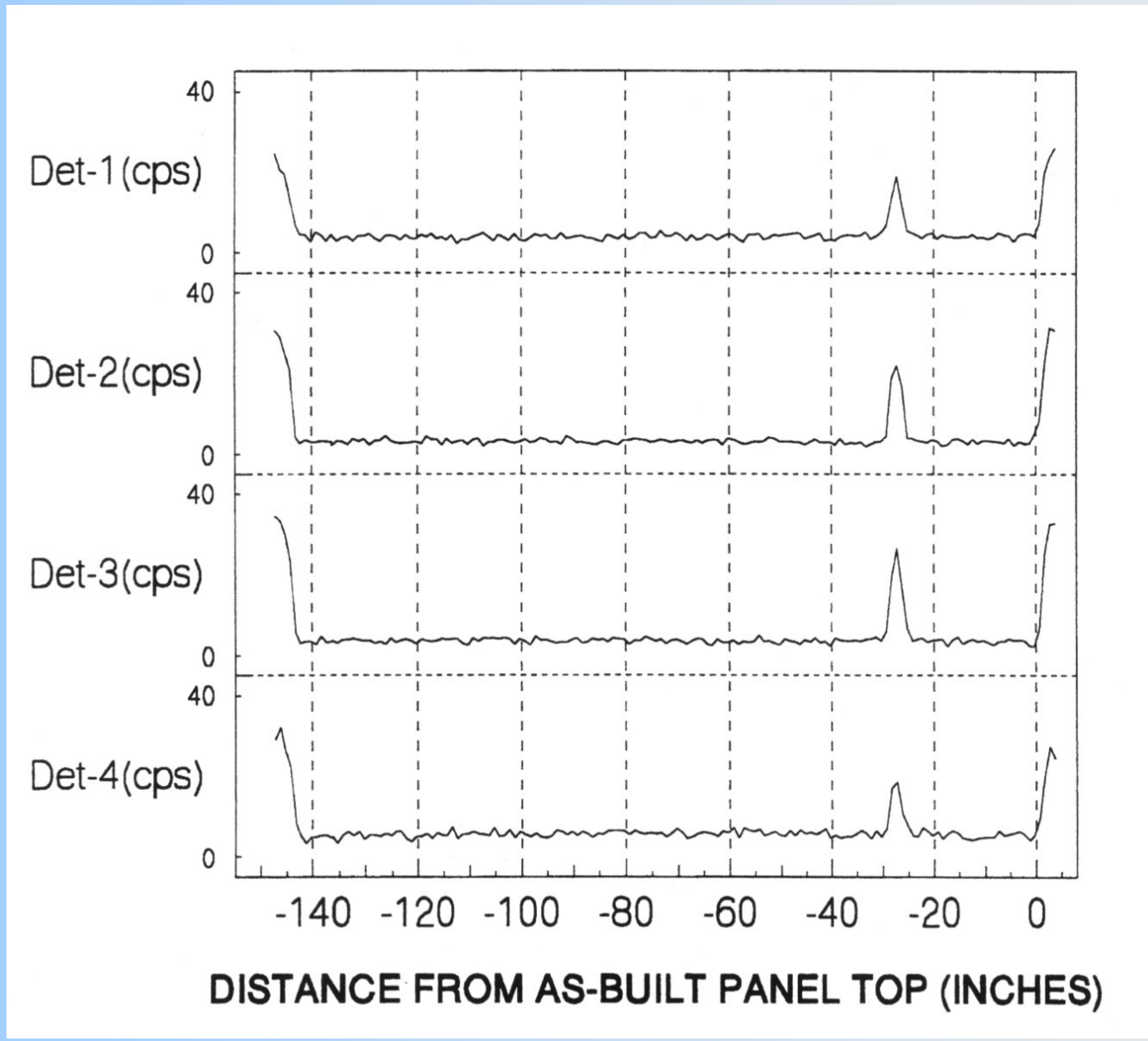
- Densification



- Shrinkage
  - Worst case: 4 inch coplanar gap in every panel at the midplane:
    $\Delta k \approx +0.04$
  - Worst case end pull back:
    $\Delta k \approx +0.0015$

NETCO

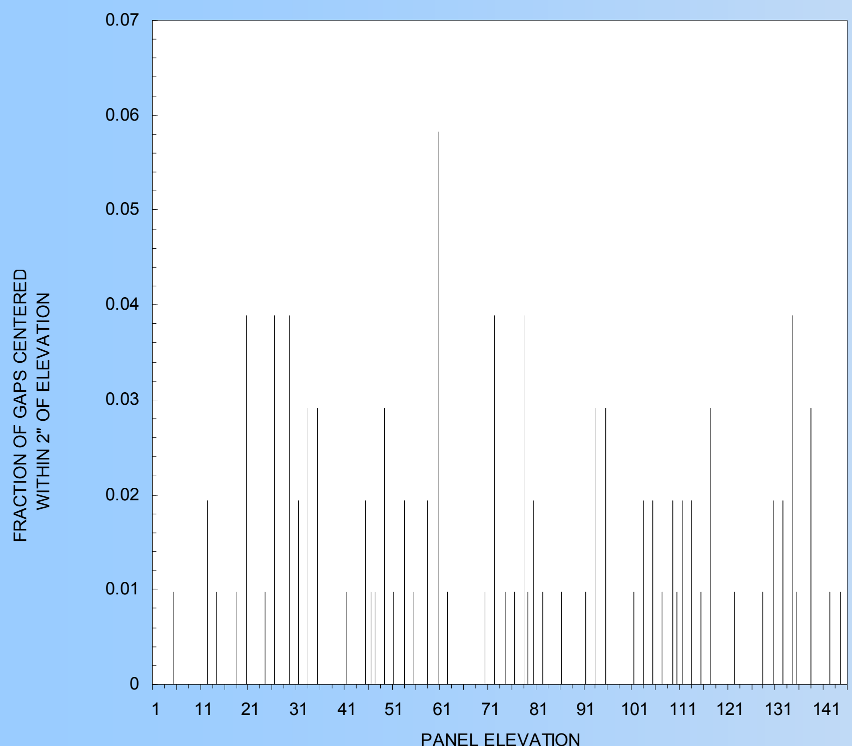
Actual Distribution of Gaps
BADGER Detector Outputs



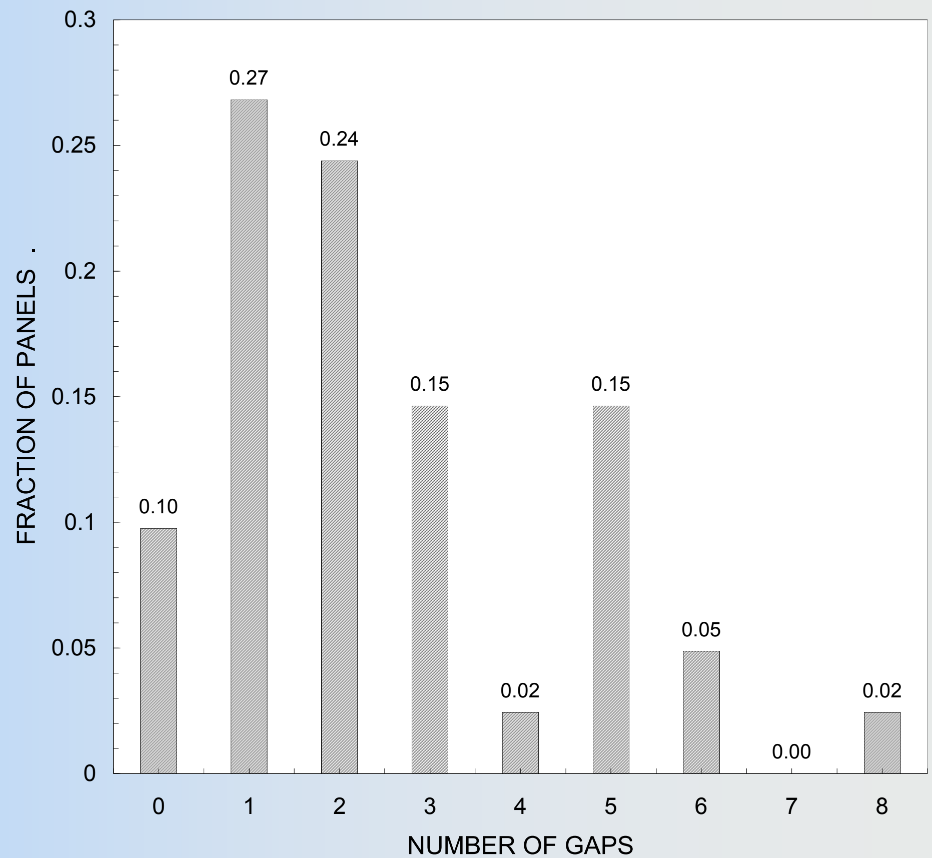
NETCO

Actual Distribution of Gaps

Axial Distribution

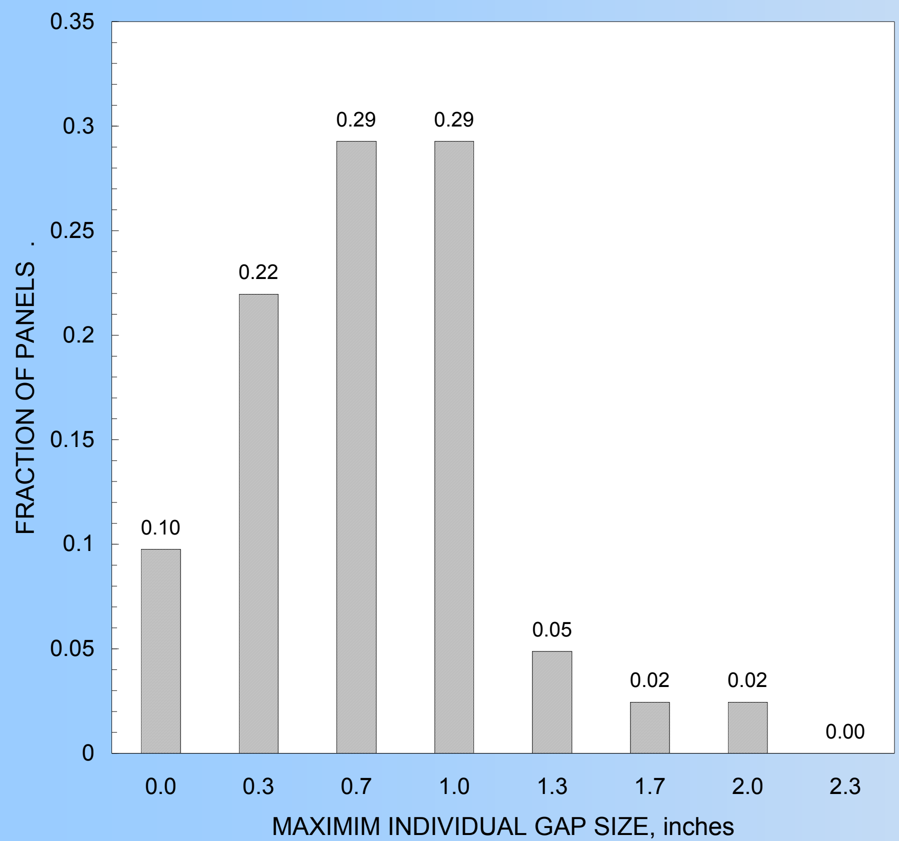


Multiple Small Gaps per Panel

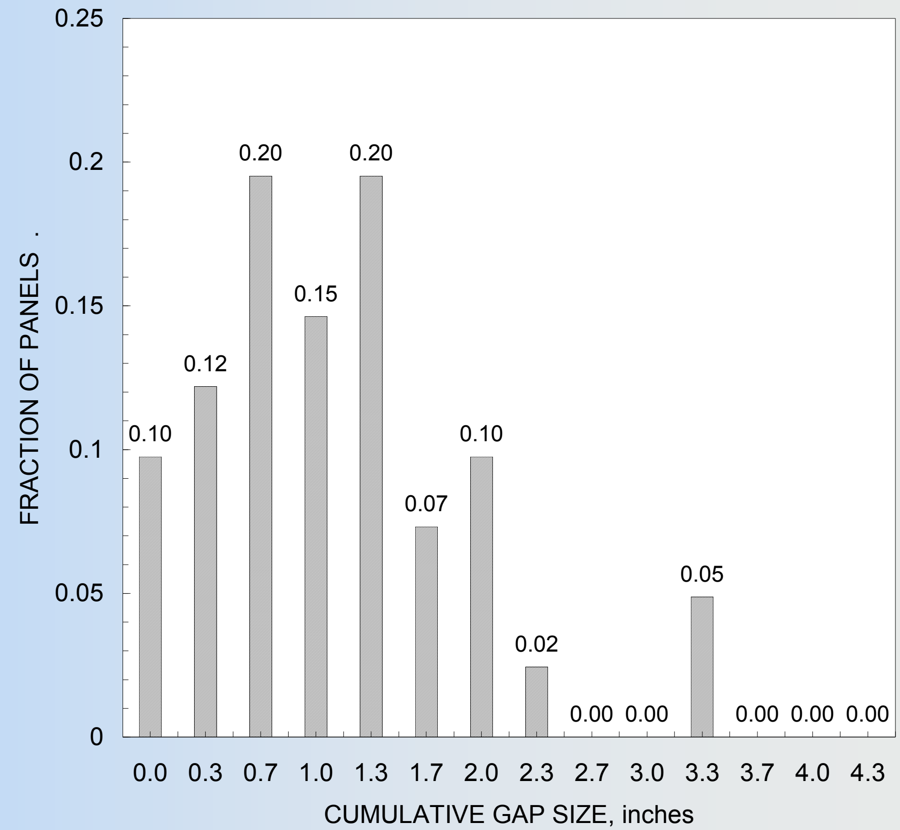


Actual Distribution of Gaps

Individual Gap Size Distribution

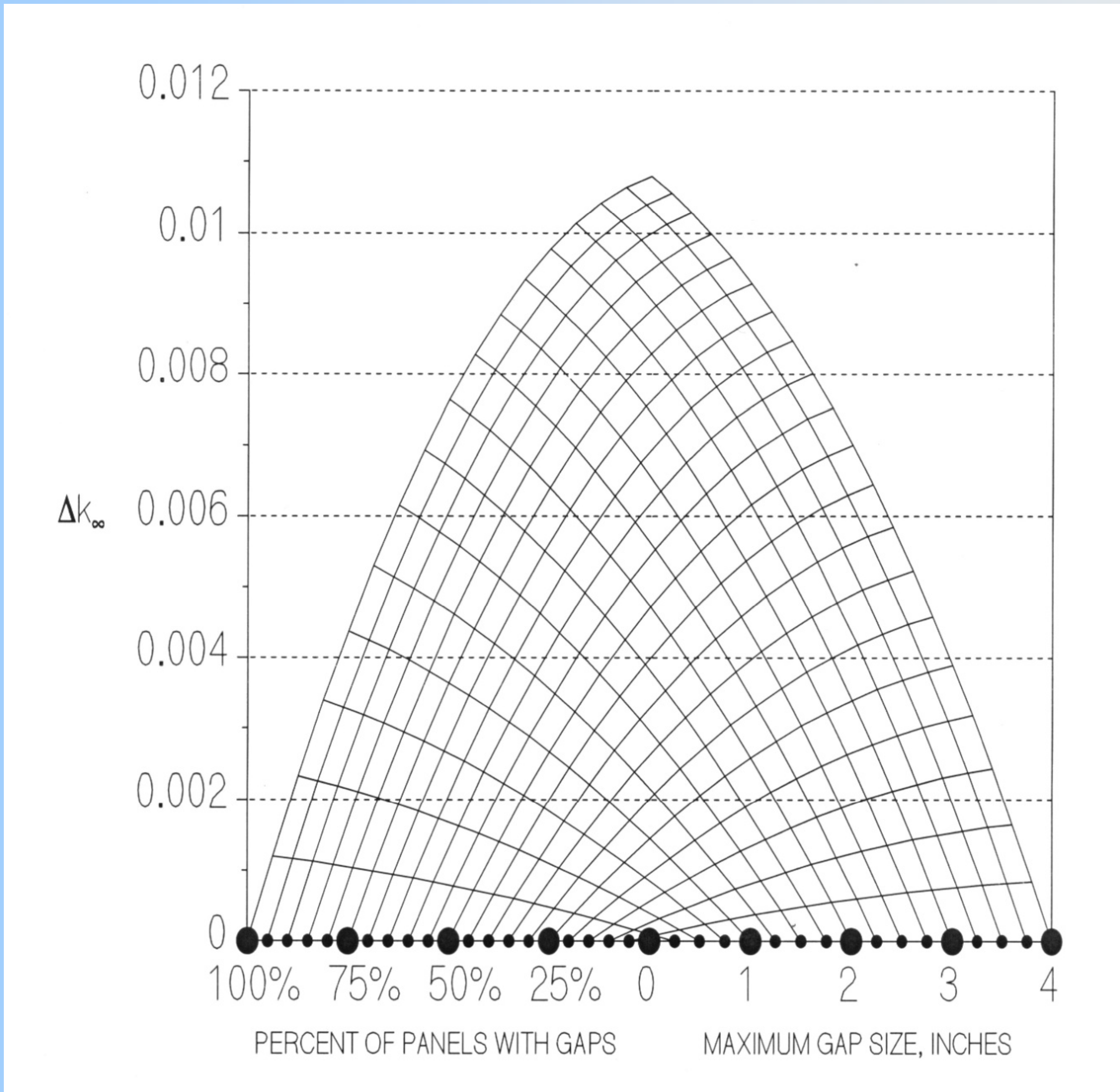


Cumulative Gap Size Distribution



NETCO

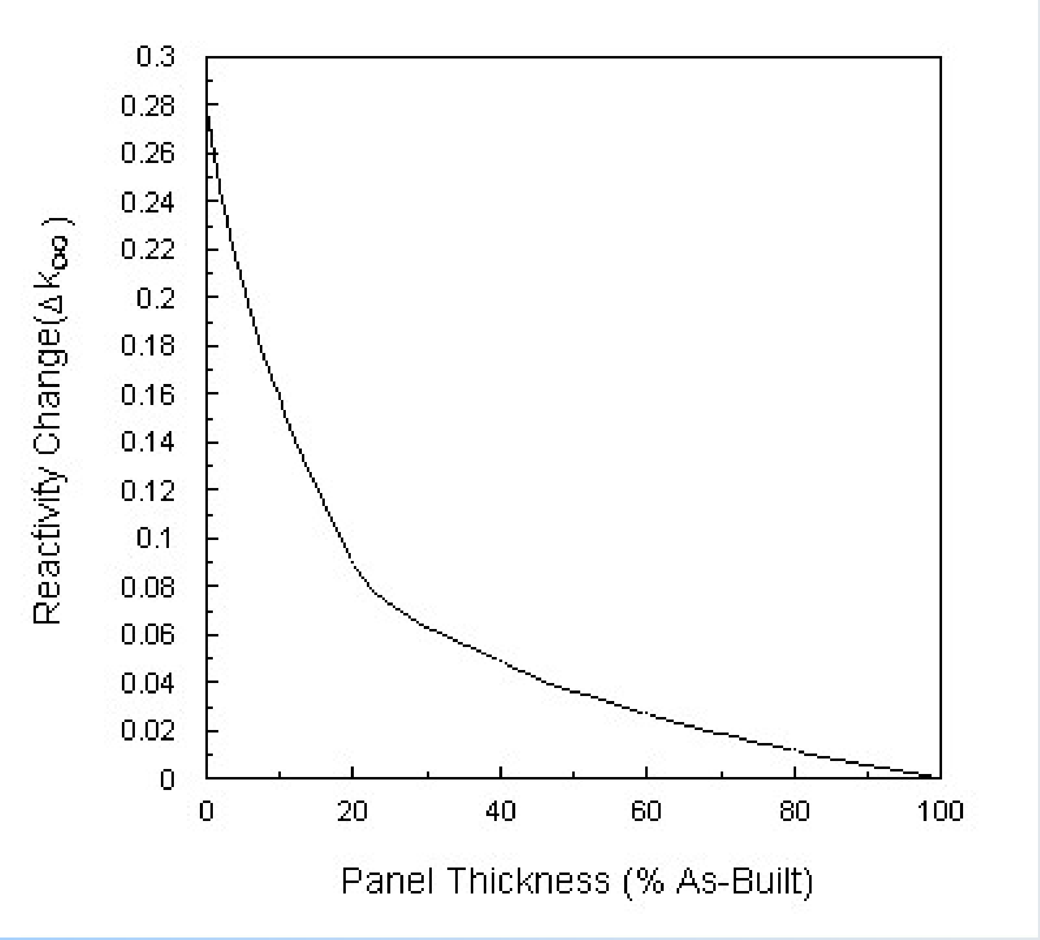
Reactivity Effect of Axially Distributed Gaps



NETCO

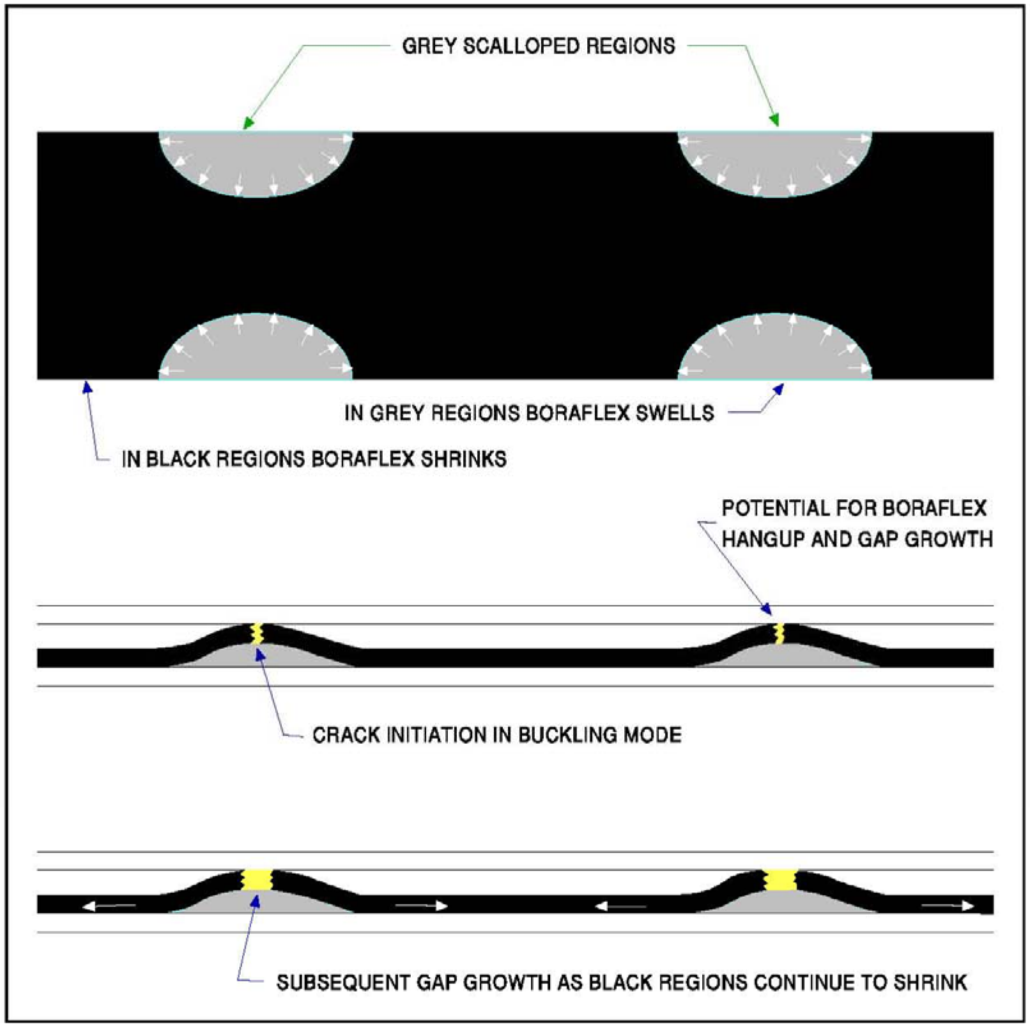
Reactivity Effects of Boraflex Dissolution

Generalized Panel Thinning



NETCO

Local Dissolution and Gap Formation

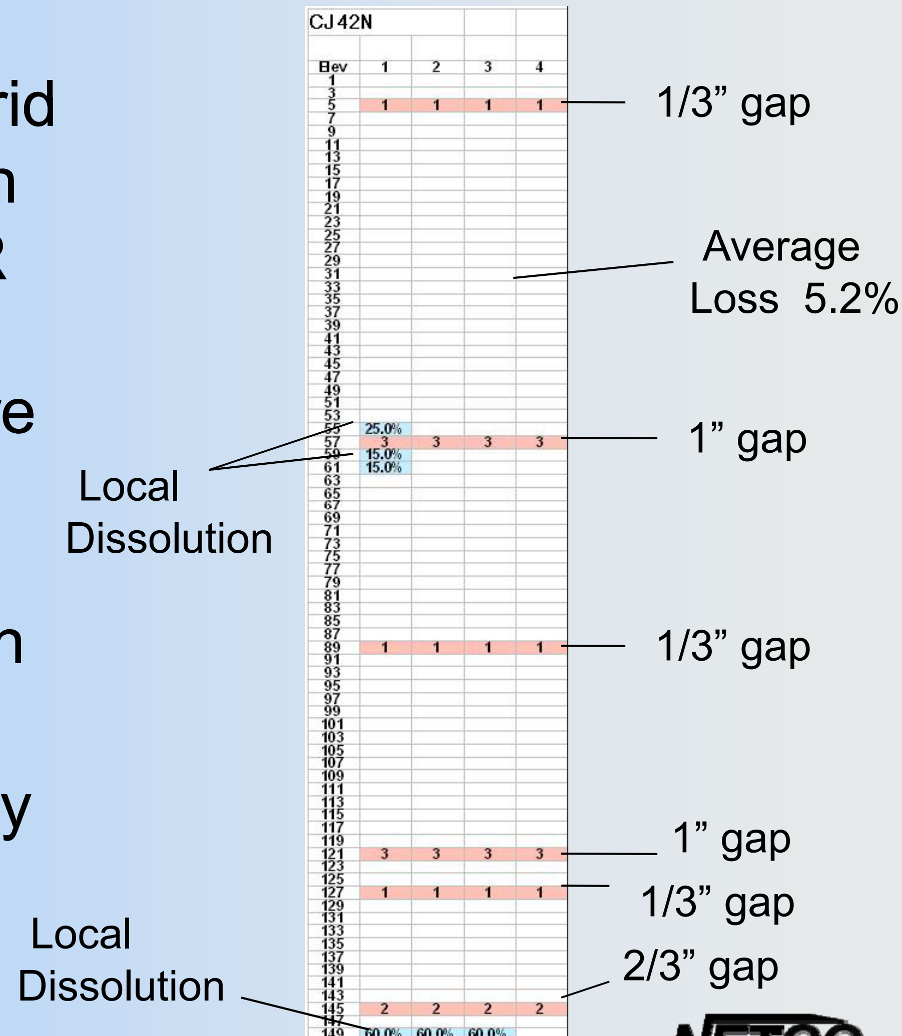


NETCO

Characterization of Panel Condition Based on BADGER Data

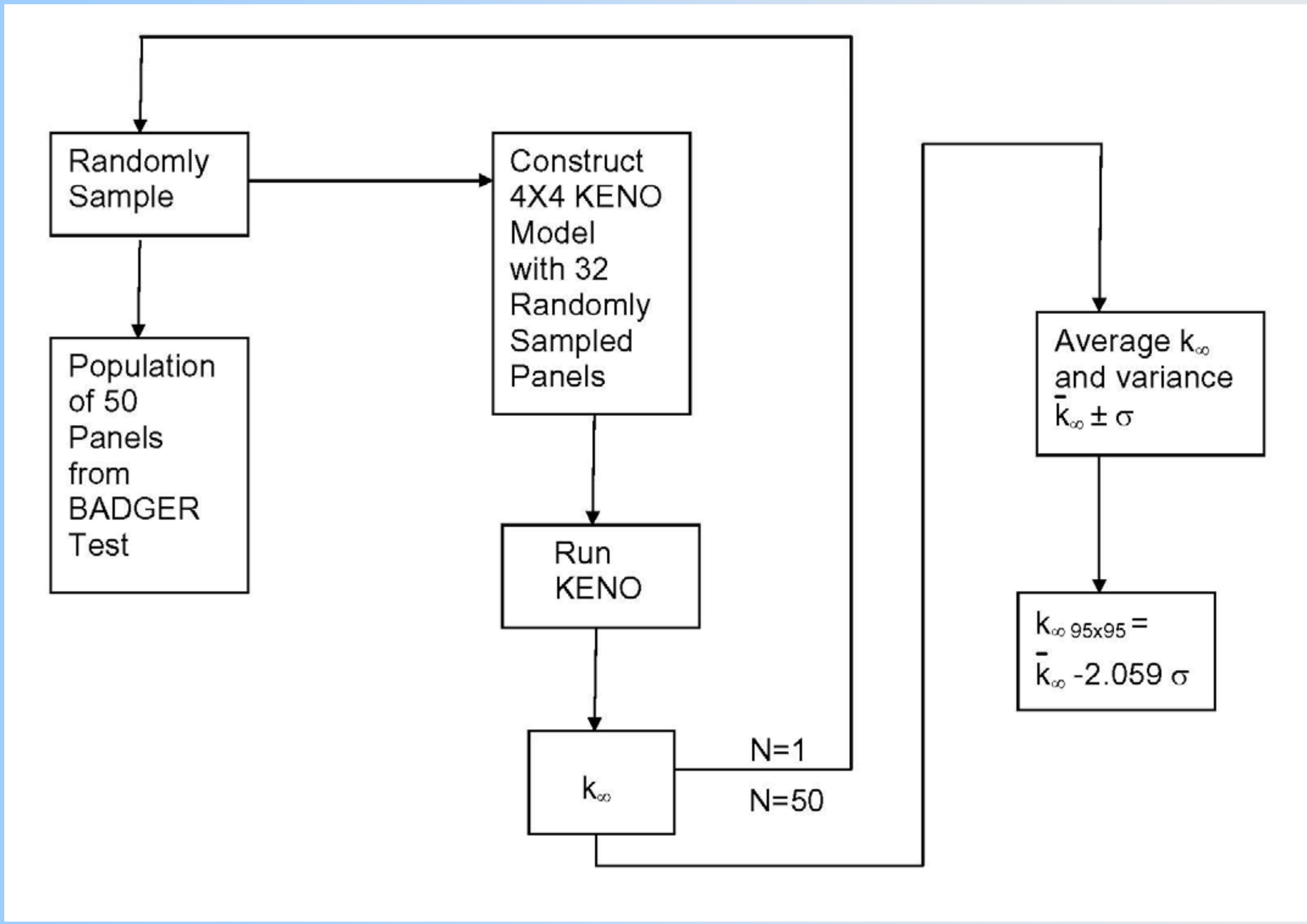
- Panel rendered as a grid 2” high and panel width divided by 4 (BADGER detectors have active region 2” high and there are 4 detectors)

- Characterize panel with respect to:
  - Average Areal Density
  - Gaps
  - Local Dissolution



NETCO

One Method of Calculation of k_∞ in Racks with Degraded Boraflex



NETCO

Reactivity Effect of Degraded Boraflex

Example of Δk_∞ Attributable to Boraflex Degradation

Boraflex Condition:	
Maximum Number of Gaps per Panel	8
Average Number of Gaps per Panel	3.6
Maximum Individual Gap Size	1.0
Average Individual Gap Size	0.4
Maximum Inches of Local Dissolution per Panel	52
Average Inches of Local Dissolution per Panel	17
Average Panel B_4C Loss	10.9%
Maximum Panel B_4C Loss	33%

Reactivity Effect:	
Degraded k_∞ (95x95)	0.938
As-Built k_∞ (95x95)	0.921
Δk_∞	0.017

NETCO

Conclusions

- After gap formation, Boraflex degradation is a gradual process characterized by local dissolution and potentially a general thinning.
- As such, boron carbide loss is distributed.
- Realistic calculation of reactivity effects of distributed boron carbide losses demonstrate that the effects are small.
- Boraflex degradation can be monitored and managed.